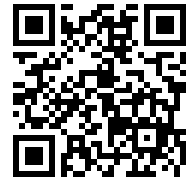

NCRP REPORT No. 38

PROTECTION AGAINST NEUTRON RADIATION

Recommendations of the
NATIONAL COUNCIL ON RADIATION PROTECTION
AND MEASUREMENTS

Issued January 4, 1971

National Council on Radiation Protection and Measurements
4201 CONNECTICUT AVENUE, N.W. / WASHINGTON, D.C. 20008

Library of Congress Catalog Card Number 73–138550

Preface

In 1957 a report entitled *Protection Against Neutron Radiation Up to 30 Million Electron Volts* was issued by the National Committee on Radiation Protection and Measurements (NCRP). That document (NCRP Report No. 20) was published as Handbook 63 of the National Bureau of Standards. In the years following the drafting of those recommendations a number of developments have taken place making it desirable to issue a new report. Most of the changes have been prompted by a new formulation of applicable quantities and units carried out by the International Commission on Radiation Units and Measurements (ICRU) and changes in the maximum permissible dose equivalents recommended by the NCRP and the International Commission on Radiological Protection (ICRP).

The present report was prepared by the Council's Scientific Committee 4 on Heavy Particles (Neutrons, Protons and Heavier). Serving on the Committee during the preparation of the report were:

H. H. Rossi, *Chairman*

Members	*Consultants*
R. S. Caswell	R. G. Alsmiller, Jr.
F. P. Cowan	L. L. Anderson
W. W. Havens	F. H. Clark
F. C. Maienschein	
L. D. Marinelli	
W. S. Snyder	
C. A. Tobias	

The Council wishes to express its appreciation to the members and consultants of the Committee for the time and effort devoted to the preparation of this report.

Lauriston S. Taylor
President, NCRP

Washington, D. C.
September 1, 1970

Contents

I. INTRODUCTION

1. Scope

In 1957 the National Committee on Radiation Protection and Measurements, the predecessor of the National Council on Radiation Protection and Measurements (NCRP), issued NCRP Report No. 20 entitled *Protection Against Neutron Radiation Up to 30 Million Electron Volts* (NCRP, 1957b).[1] In the years following the drafting of the recommendations set out in that report, a number of developments have taken place making it desirable to issue a new report. Calculations have been made of depth dose distributions for neutron energies up to 400 MeV. Also, new shielding calculations have become available. There have not been many further developments in the dosimetry of neutrons and mixed radiations following the publication in 1961 of NCRP Report No. 25, *Measurement of Absorbed Dose of Neutrons and of Mixtures of Neutrons and Gamma Rays* (NCRP, 1961), but certain practical improvements have occurred, particularly in the area of devices that give a direct estimate of dose equivalents.

In addition to the elimination of an explicit upper energy limit, the present report differs from its predecessor, NCRP Report No. 20, in that this report includes (1) a formulation of permissible values of neutron fluence for energies up to 14 MeV, based on the distribution of absorbed dose in truncated cylindrical phantoms rather than in infinite slabs; (2) corresponding data for slabs in the energy range from 0.5 to 400 MeV; and (3) more specific shielding data. Discussion of biological effects of neutrons has been omitted because background information on this subject may now be obtained elsewhere. There is a considerable number of lesser changes dictated by new information from a variety of sources.

As in the earlier report (NCRP No. 20), the scope of this report is primarily restricted to considerations arising in routine operation of various neutron sources. No reference is made to design or operating procedures of reactors or other critical assemblies, although the discussions of routine and accident dosimetry are applicable to these devices.

[1] Literature references are listed on page 132.

2. General Principles of Neutron Protection

2.1 Neutrons constitute the most important radiation for which protection considerations must take into account not only radiation quantity but also radiation quality. X and gamma rays of energies in common use produce substantially equal biological effects for equal doses. However, the biological effectiveness of neutrons usually is not only higher but also depends markedly on neutron energy.

2.2 It is assumed that differences in the biological effects of radiations are related to differences in linear energy transfer (LET) of the charged particles that deliver the absorbed dose. Consequently, the limits of radiation exposure of personnel are expressed in terms of the dose equivalent (DE), which is defined as the product of absorbed dose (D) and the quality factor (QF).[2] The latter factor is specified as a function of LET (NCRP, 1954; ICRP/ICRU, 1963). The quality factor is unity for x and gamma rays. In all practical cases involving more densely ionizing particles, the dose is delivered over a range of values of LET (L) and in this case the dose equivalent is given by

$$\mathrm{DE} = \int_{L_{\min}}^{L_{\max}} D(L)\ \mathrm{QF}\ (L)\ \mathrm{d}L \qquad (1)$$

where $D(L)$ is the distribution of absorbed dose in LET, L, and $\mathrm{QF}(L)$ is the quality factor at L.

2.3 In principle, protection measurements in an unknown radiation field require a determination of $D(L)$ (Rossi et al., 1962). A method for the experimental determination of this function exists, but, because of its complexity, it is infrequently employed. The approach usually chosen instead is to employ procedures which discriminate between doses delivered by various radiations and to apply a conservatively chosen value of the quality factor to each such dose.

[2] The term QF has replaced RBE and the term dose equivalent has replaced RBE dose. These changes were recommended because RBE, as used in radiobiology, represents a measurable ratio that applies only in a well-defined set of circumstances, while what is now termed QF is a factor which is designed to apply to radiation protection provided absorbed doses are less than of the order of 10 rads. Definitions of these terms, and others used in this report, are set out in Appendix A.

2.4 In virtually all protection surveys it will be found that neutrons are accompanied by gamma radiation. For neutrons of energies up to at least 15 MeV in such a mixed field, one may determine the absorbed dose due to gamma rays and that due to neutrons separately, multiply the former by 1 and the latter by 10, and add these products to obtain a conservative estimate of the dose equivalent.

2.5 A still simpler procedure is to determine the total absorbed dose and apply a quality factor of 10, which results in a conservative assessment of the dose equivalent regardless of the neutron to gamma dose ratio.

II. ELEMENTS OF NEUTRON PROTECTION

3. Classification of Neutrons and Primary Modes of Interaction

3.1 Neutrons are produced by various devices such as nuclear reactors and accelerators in an energy range (per particle) from about 10^{-4} to 10^{10} eV, and are found at still higher energies in cosmic radiation. The types of interaction vary markedly with energy. It is convenient to classify neutrons roughly according to energy ranges where certain interactions predominate. The transition between ranges is not sharp, leaving considerable latitude in the choice of limits.

3.2 Thermal neutrons are neutrons in approximate thermal equilibrium with matter. The distribution of velocities is approximately Maxwellian although the temperature characteristic of this distribution may often be higher than that of the medium. "Cold" neutrons available from special facilities within a nuclear reactor may have a Maxwellian distribution characteristic of a temperature below room temperature, while some nuclear reactors produce thermal neutrons with a characteristic temperature considerably above room temperature. For a temperature of 20.44°C, the most probable velocity in the distribution is 2,200 meters per second, corresponding to an energy of 0.0253 eV. In many instances it is sufficiently accurate to consider thermal neutrons as monoenergetic with this energy. Generally the energies associated with thermal neutrons are below about 1 eV, the limit depending on the temperature considered. One may choose an energy above which the relative number of neutrons in the Maxwellian distribution becomes very small, or the "cadmium cutoff" energy of about 0.5 eV (since the "cadmium difference method" is often used in thermal neutron measurements). In quantitative measurements of thermal neutrons it is necessary to be clear on definitions of the quantities being measured. For this purpose the reader is referred to Section 9. For radiation protection the most important interaction of thermal neutrons with matter is ra-

diative capture. In this process the neutron is captured by the nucleus with the emission of gamma radiation. In tissue irradiated by thermal neutrons, elastic scattering occurs more frequently than capture, but is less important because neither the neutrons nor the matter traversed gain or lose energy on the average. Occasionally, nuclear reactions such as (n,p) or (n,α) or fission may occur. The $^{14}N(n,p)^{14}C$ reaction is important in tissue.

3.3 The intermediate energy region is associated with peaks or resonances in the interaction cross section for neutrons and typically is considered as extending from the electron volt (eV) region of energies to tens of keV. Intermediate neutrons are usually obtained from the moderation of fast neutrons by elastic collisions. This slowing-down process is the most important interaction between intermediate neutrons and matter, and it typically leads to a neutron flux density that is inversely proportional to energy—the "dE/E spectrum". Capture and nuclear reactions may also occur. The resonances in the neutron cross sections in this energy region, while very important for many materials, are relatively unimportant for tissue.

3.4 Fast neutrons are neutrons of energy higher than that of intermediate neutrons. Up to energies of the order of 10 MeV, the most important interaction with matter is elastic scattering. At energies of the order of 1 MeV inelastic scattering can become appreciable. Nuclear reactions become important for neutrons of several MeV. In tissue, and for neutron energies in excess of 10 MeV, inelastic scattering and nuclear reactions (frequently with the emission of several particles) become comparable in frequency with elastic scattering.

3.5 The probability of any interaction between neutrons and matter is expressed quantitatively in terms of cross sections. The cross section, σ, can be considered to be the effective target area of the nucleus if the neutron is assumed to have zero diameter. For a beam containing n neutrons per unit volume moving with velocity v toward N nuclei, the number of interactions per unit time will be $nv\mathrm{N}\sigma$. The quantity nv is the neutron flux density. The cross section is usually expressed in barns (10^{-24} cm^2).

3.6 Each possible nuclear interaction can be characterized by a cross section. In addition, cross sections for scattering processes may be further divided into differential cross sections that express the probability for scattering in particular directions or into particular energy ranges. The number of neutrons interacting is determined by all processes that can take place, that is, by the total neutron cross section, σ_t.

4. Radiation Quantity

4.1 The energy imparted to tissues has been adopted as a principal physical basis for quantitative correlation between irradiation and biological effect. The energy per unit mass that is imparted to matter by ionizing radiation is the absorbed dose. Its special unit is the rad. The dose equivalent for any organ is equal to the absorbed dose received by that organ times the applicable $\overline{QF}$.

4.2 Kerma, although often approximately equal to absorbed dose, is a distinctly separate concept. The difference may be illustrated by consideration of a beam of neutrons incident normally on a uniform slab of tissue. The tissue kerma, which is a measure of the ability of the neutron radiation to transfer kinetic energy to secondary charged particles such as proton and carbon recoils, generally decreases with increasing depth in the slab as the primary neutron radiation is attenuated. On the other hand, the absorbed dose, which is a measure of the energy imparted to matter by the various recoils, first increases with depth as the total number of recoils ejected by neutrons builds up, goes through a maximum, and then decreases as the primary neutron radiation is attenuated. In those regions of the slab which are beyond the transition zone (i.e., beyond the maximum of the absorbed dose), tissue kerma and absorbed dose are usually very nearly numerically equal, since the recoil ranges are short relative to distances in which the primary radiation is attenuated appreciably.

4.3 If other quantities (such as flux density) are utilized to specify radiation quantity, subsidiary information must be supplied to permit derivation of the absorbed dose. However, some kind of derivation must nearly always be performed because it is usually not practical to determine the dose received by organs of an individual or in corresponding locations within a phantom. The procedure commonly employed instead is to perform a measurement in "free space". The measurement may be made utilizing a tissue equivalent ionization chamber, proportional counter, etc. with a wall thickness just sufficient to establish radiation equilibrium. The quantity evaluated under these conditions is approximately the tissue kerma in free space or alternatively the absorbed dose to the mass of the tissue equivalent material in the sensing portion of the detector. A different kind of determination may be based on a measurement of neutron fluence in free space. In either case, the evaluation of a dose which would have been received at depth

in an individual at the location in question requires consideration of a number of factors such as neutron energy distribution, absorption, scattering, and the mass energy transfer coefficient. Information on these factors will be given in pertinent sections of this report. However, procedures involving many physical factors often result in substantial uncertainties in the dose equivalent. Assessments employing a minimum of conversion factors are therefore likely to be more accurate.

5. Radiation Quality

5.1 The definition of the dose equivalent given in equation 1 (Section 2.2) may be used as a basis for the further definition of a mean quality factor $\overline{\mathrm{QF}}$ as follows:

$$\overline{\mathrm{QF}} = \frac{\mathrm{DE}}{D} = \frac{\int_{L_{\min}}^{L_{\max}} D(L)\mathrm{QF}(L)\,\mathrm{d}L}{\int_{L_{\min}}^{L_{\max}} D(L)\,\mathrm{d}L} \qquad (2)$$

$\overline{\mathrm{QF}}$ is a single factor by which the absorbed dose is multiplied in order to obtain the dose equivalent. Values of $\overline{\mathrm{QF}}$ for various neutron energies are given in Table 2 (Section 7.4). It will be noted that in the energy range covered by the table, $\overline{\mathrm{QF}}$ does not exceed 10 except for a minor excursion at 1.0 MeV which may be neglected. This fact forms the basis of a commonly employed conservative procedure in which the neutron dose is multiplied by a factor of 10 in order to obtain its contribution to the dose equivalent.

5.2 As shown below (Appendix B.I.) the $\overline{\mathrm{QF}}$ is also a function of depth of penetration in tissue for neutron energies up to 14 MeV. However, the surface value is virtually never exceeded at greater depths.

5.3 The evaluation of the dose equivalent (DE) may be performed on the basis of an assessment of the absorbed dose, as outlined in Section 4.3, and the selection of a suitable value of $\overline{\mathrm{QF}}$. The latter may be based on any of the approaches enumerated in Sections 2.3, 2.4 and 2.5.

An alternate method is the employment of devices having a response that is proportional to the calculated DE at some location in a phantom of typical dimensions irradiated in a standard manner. Usually the point of interest has been taken as that where the maximum dose equivalent occurs in a 30-cm slab irradiated by a perpendicularly incident broad beam, but some of the calculations presented below are based on a cylindrical phantom (Anderson and Braun, 1964; Dennis and Loosemore, 1960; Hankins, 1963; and Leake, 1967). If the proportionality factor between instrument reading and dose equivalent is substantially independent of neutron energy over the range subject to measurement, such devices are useful although their accuracy is limited if only because of questions of geometry. Since directional characteristics would compound the uncertainty, such detectors are usually de-

signed to have an isotropic response. Consequently they yield the same reading whether irradiated isotropically or unilaterally. On the other hand, an individual located at the position of the detector could receive quite different dose equivalents under these two conditions. The reading of the instrument is, therefore, at best an upper limit of the maximum dose equivalent to be expected.

6. Interactions Between Neutrons and Tissue

6.1 A wide variety of interactions of neutrons with elements present in tissue is possible. The elemental compositions of various types of tissues are given in Table 1 (Tipton et al., 1969). The "skeleton" includes bone as well as marrow and other organic components present in a freshly dissected skeleton. The term "bone" represents mineralized bone.

The following is a summary of the principal physical characteristics of interactions between neutrons and elements in tissue. Detailed numerical data are given in Appendix B.

6.1.1 *Elastic Collision.* These interactions are those in which the neutron and the nucleus may be assumed to retain their identity and kinetic energy is conserved. If θ is the neutron scattering angle in the center of mass system of coordinates, the neutron energy loss is given by

$$\Delta E_n = 2\,MmE_0\,(1 - \cos\theta)/(M + m)^2$$

where E_0 is the neutron energy before the collision and M and m are the masses of the struck particle and of the neutron, respectively. If the recoil nucleus moves in the laboratory system at angle $\psi = (\pi - \theta)/2$ relative to the initial velocity vector of the neutron, this becomes

$$\Delta E_n = \frac{4MmE_0}{(M + m)^2}\cos^2\psi$$

For most of the elements comprising tissue, the cross-section for elastic scattering of a neutron with the element is well known. Below 10 MeV elastic scattering with hydrogen is nearly isotropic in the center of mass system of coordinates but it is known to be anisotropic for the other elements at most energies. Nevertheless, calculations based on the assumption that elastic scattering is isotropic have not shown a marked departure from those using anisotropy so far as dose in an anthropomorphic phantom is concerned and for neutron energies not exceeding 14 MeV. This is not surprising in view of the preponderance of hydrogen atoms in body tissues and the fact that below this energy 90 percent or more of the dose is due to elastic scattering with hydrogen or to capture processes where isotropy also may be assumed.

TABLE 1—*Atomic composition of tissue*

	Whole Body		Adipose Tissue		Muscle		Bone[a]		Skeleton[b]	
	% by wt.	atoms/g	% by wt.	atoms/g	% by wt.	atoms/g	% by wt.	atoms/g	% by wt.	atoms/g
Oxygen	61	2.29×10^{22}	23	8.66×10^{21}	75	2.82×10^{22}	43	1.62×10^{22}	49	1.84×10^{22}
Carbon	23	1.15×10^{22}	64	3.21×10^{22}	11	5.52×10^{21}	16	8.02×10^{21}	23	1.15×10^{22}
Hydrogen	10	6.02×10^{22}	12	7.23×10^{22}	10	6.02×10^{22}	4.1	2.47×10^{22}	7.1	4.27×10^{22}
Nitrogen	2.6	1.12×10^{21}	0.80	3.44×10^{20}	2.6	1.12×10^{21}	4.3	1.85×10^{21}	3.9	1.68×10^{21}
Calcium	1.4	2.11×10^{20}	0.0022	3.31×10^{17}	0.0031	4.66×10^{17}	21	3.16×10^{21}	10	1.50×10^{21}
Phosphorus	1.0	1.94×10^{20}	0.016	3.10×10^{18}	0.18	3.50×10^{19}	10	1.94×10^{21}	7.0	1.36×10^{21}
Sulfur	0.20	3.76×10^{19}	0.073	1.37×10^{19}	0.23	4.32×10^{19}	0.31	5.82×10^{19}	0.17	3.19×10^{19}
Potassium	0.20	3.08×10^{19}	0.032	4.87×10^{18}	0.30	4.62×10^{19}			0.15	2.29×10^{19}
Sodium	0.14	3.67×10^{19}	0.050	1.31×10^{19}	0.075	1.96×10^{19}	0.62	1.60×10^{20}	0.32	8.38×10^{19}
Chlorine	0.12	2.04×10^{19}	0.12	2.04×10^{19}	0.078	1.33×10^{19}			0.14	2.37×10^{19}
Magnesium	0.027	6.69×10^{18}	0.0020	4.95×10^{17}	0.019	4.68×10^{18}	0.22	5.45×10^{19}	0.12	2.97×10^{19}
Total	99.687		99.92		99.48		99.55		100.9	

[a] Mineralized bone.

[b] Includes bone, bone marrow, and other organics present in the intact skeleton as dissected.

6.1.2 *Inelastic Scattering.* The term inelastic scattering is used here to designate only those interactions in which the neutron may be considered to undergo transitory capture and re-emission. The struck nucleus is left in an excited state, and one or more photons, frequently of high energy, are emitted, usually within a fraction of a second. This type of interaction is not possible unless the neutron energy exceeds a certain threshold necessary to conserve both energy and momentum. The threshold energy, E_{thresh} is given by

$$E_{\text{thresh}} = E_\gamma \frac{M + m}{M}$$

where $E\gamma$ is the energy of the gamma ray photon.

6.1.3 *Capture Processes.* These processes are defined here as those in which the nucleus absorbs a neutron of low energy and emits a photon. The cross sections for many of these capture processes have a $1/v$ dependence (where v is the velocity of the neutrons). The $^1H(n,\gamma)^2H$ reaction is the most important in tissue. It produces a photon of 2.2 MeV. For neutrons of low energy, this interaction contributes much of the dose throughout the human body and for neutron energies up to 2.5 MeV, it contributes a major portion of the dose in deep layers, since the photons can penetrate over considerable distances prior to electron interactions.

6.1.4 *Nuclear Interactions at Lower Energies (<20 MeV).* As used here, this term is limited to those reactions where protons, or other heavier particles are emitted. For thermal neutrons the $^{14}N(n,p)^{14}C$ interaction is of particular importance in tissue. It produces a proton of 0.58 MeV energy. In interactions initiated by neutrons of higher energy, the reaction products may have various energies, depending on both the energy level involved and the angle of emission. In most cases the applicable data are not known with sufficient accuracy to make a precise determination of the average energy of the reaction products.

6.1.5 *Nuclear Reactions at Higher Energies (>20 MeV).* Nuclear reactions at the higher neutron energies are qualitatively not different from nuclear reactions at the lower neutron energies. Quantitatively, however, the number of particles emitted per reaction increases as the neutron energy increases, and therefore nuclear reactions are very important in the dosimetry of the higher energy neutrons. Experimental data on the energy and angular distribution of particles emitted from high-energy reactions are very limited, and the detailed information needed for dosimetry calculations in general must be obtained from theoretical considerations.

For neutrons having energies between 50 and 100 MeV, reactions

may be thought of as consisting of three stages. In the first stage, a cascade develops inside the nucleus. The incident high-energy neutron interacts with an individual nucleon in the nucleus, and the scattered and the recoiling nucleons from this interaction proceed through the nucleus. Each of these nucleons may in turn interact with other nucleons in the nucleus, and thus a cascade develops. Some of these "cascade" particles have sufficiently high energy that they escape from the nucleus while some of them are of low energy and do not escape. After the fast particles have escaped, the energy of the slow particles is assumed to be distributed among all of the remaining nucleons in the nucleus, and one has a residual nucleus in an excited state. In the second stage of the reaction, this residual nucleus evaporates particles. In this evaporation process, heavy nuclei such as alpha particles (which are important because of their high LET) as well as nucleons are emitted. Finally, in the third stage of the reaction, after particle emission is no longer energetically possible, the remaining excitation energy is emitted in the form of γ rays.

In the energy region below 50 MeV, this three-stage description of a nuclear reaction is inappropriate, and, as an alternative, it is sometimes assumed that only the last two stages are operative, i.e., that the energy of the incident neutron is distributed among all the nucleons in the nucleus before particle emission begins and that all emitted particles arise by evaporation from the excited nucleus. For neutron energies near 50 MeV neither description is entirely appropriate, and it is not known which is more accurate (see Appendix B.II).

6.2 Sufficient numerical data are available on the interactions discussed in Section 6.1 to permit calculations of the tissue kerma in free space as well as of the depth-distribution of both absorbed dose and dose equivalent in tissue equivalent phantoms for a wide range of neutron energies (see Appendix B).

7. Maximum Permissible Dose Equivalent

7.1 Because of the action of cosmic radiation, there exists a rather constant neutron flux density of about 60 cm^{-2} h^{-1} at sea level. This increases with altitude, reaching a value of approximately 600 cm^{-2} h^{-1} at 10,000 feet. The resultant tissue kerma rate in free air is of the order of 10^{-3} erg g^{-1} $week^{-1}$ at sea level and 10^{-2} erg g^{-1} $week^{-1}$ at 10,000 feet. The approximate spectral distribution at sea level is shown in Figure 1 (Hess et al., 1959).

7.2 The maximum permissible dose equivalent is a dose equivalent that may be received without undue risk to the health of the individual and that of the population. Certain basic rules are given in other NCRP publications: NCRP Report No. 17 (NCRP, 1954), NCRP Statement

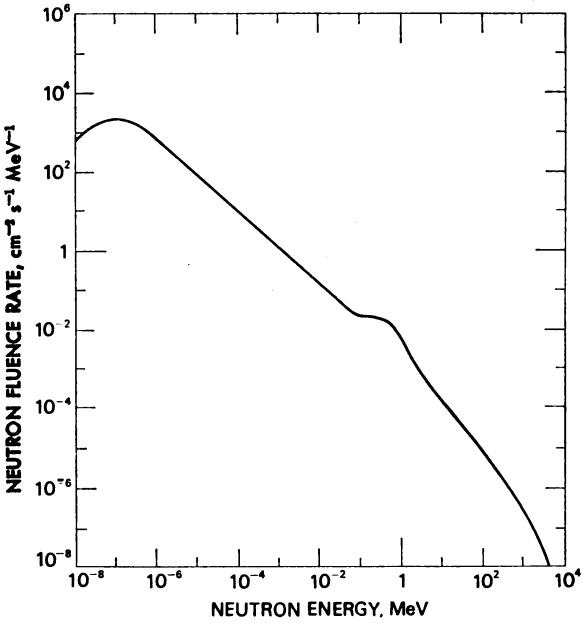

Fig. 1. Cosmic ray neutron energy spectrum at sea level. (From Hess, W. N., Patterson, H. W., Wallace, R. and Chupp, E. L., "Cosmic-ray Neutron Energy Spectrum," Phys. Rev. **116**, 445, 1959).

(NCRP, 1957a), and NCRP Addendum (NCRP, 1958); and in a new report now in preparation (NCRP, 1971). The present rules state that for radiation workers, the dose equivalent to the whole body, head and trunk, active blood-forming organs or gonads *shall not* exceed 3 rems in any 13 consecutive weeks and that the accumulated dose equivalent at any age *shall not* exceed 5 rems multiplied by the number of years beyond age 18. (See Appendix A for the special significance of the word *shall* as used in this report.) In addition, limits of 10 rems in 13 weeks and 30 rems per year are specified for the skin of the whole body (NCRP, 1960b). Limits for hands and forearms or feet and ankles are 25 rems in 13 weeks and 75 rems per year.

Recent recommendations of the NCRP also take cognizance of the potential exposure of the fetus resulting from the occupational exposure of pregnant and potentially pregnant women. They specify that during the entire gestation period the dose equivalent to the fetus from occupational exposure of the mother *should not* exceed 0.5 rem. (See Appendix A for the special significance of the word *should* as used in this report.)

For energies up to at least 400 MeV, dose equivalents of neutrons are for practical purposes always maximal at or near the body surfaces; and because some of the critical organs, such as lenses of the eyes and the male gonads, are near the body surface, these rules require that in this energy range and for whole-body exposure the limits stated above be applied at or near the body surface.

7.3 It will be noted that in the case of a radiation worker who, beginning at an early age, is routinely exposed to a substantially constant radiation level, the maximum yearly dose equivalent *shall not* exceed 5 rem, corresponding to an average weekly dose equivalent of 100 mrem. It will sometimes be necessary or convenient to plan or administer radiation protection on the basis of 100 mrem per week even though there is no longer an explicit limit on weekly dose equivalent. For this reason, certain data given below (e.g., Table 2) are based on a dose equivalent value of 100 mrem.

7.4 NCRP Report No. 17 (NCRP, 1954) contains a listing of applicable QF values according to the specific ionization of the particles delivering the dose. Calculations taking into account the LET of secondary recoils arising from both primary and multiply scattered neutrons indicate that in a phantom 30 cm thick, the $\overline{QF}$ deduced from NCRP Report No. 17 data depends both on neutron energy and, to some extent, on the depth in the phantom. However, in general, the highest $\overline{QF}$ *shall* be applied for purposes of protection. Appendix B contains figures showing the absorbed dose and the dose equivalent as a function of

TABLE 2—*Mean quality factors, $\overline{QF}$[a], and values of neutron flux density which, in a period of 40 hours, results in a maximum dose equivalent of 100 mrem*

Neutron Energy		$\overline{QF}$	Neutron Flux Density
MeV			$cm^{-2}\ s^{-1}$
2.5×10^{-8} (thermal)		2	680
1×10^{-7}		2	680
1×10^{-6}		2	560
1×10^{-5}		2	560
1×10^{-4}		2	580
1×10^{-3}		2	680
1×10^{-2}		2.5	700
1×10^{-1}		7.5	115
5×10^{-1}		11	27
1		11	19
2.5		9	20
5		8	16
7		7	17
10		6.5	17
14		7.5	12
20		8	11
40		7	10
60		5.5	11
1×10^{2}		4	14
2×10^{2}		3.5	13
3×10^{2}		3.5	11
4×10^{2}		3.5	10

[a] Maximum value of $\overline{QF}$ in a 30-cm phantom.

depth for a number of neutron energies.[3] Based on these calculations, Table 2 gives $\overline{QF}$ and maximum permissible average neutron flux density as a function of neutron energy for protracted exposure on the basis of a 40-hour week. Although a $\overline{QF}$ of 11 might be slightly exceeded at some neutron energies in the neighborhood of 1 MeV, it is sufficiently safe to derive flux densities for protection planning for any neutron energy between thermal levels and 400 MeV by linear interpolation between neighboring energies in Table 2.

7.5 It must be realized that the values in Table 2 apply only to monoenergetic neutrons incident normally on the major portions of the human body. Even when a neutron generator emits monoenergetic neutrons, scattering by walls and other structures will cause degradation in energy. However, since this process usually leads to a decreased dose equivalent per unit fluence, it is safe to assume that all neutrons have the original maximum energy.

[3] The NCRP is indebted to W. S. Snyder, R. G. Alsmiller, Jr., and their colleagues for having carried out the computations on which these figures are based.

7.6 In case sufficiently detailed information on neutron energy is not available, a $\overline{QF}$ of 10 *shall* be assumed.

7.7 The maximum dose equivalent of 12 rem in one year *shall* be permitted only when adequate past and current exposure records exist so that it can be established that such a dose will not cause the individual to exceed his age prorated allowance.

7.8 When feasible, it is desirable to design neutron installations so that an hourly rate of 2½ mrem/h will not be exceeded in normally occupied areas, as it is then impossible for most personnel to receive 5 rem in a year if they work 40 hours per week. In areas where the dose equivalent rate is in excess of 2½ mrem/h, the occupancy periods of personnel or the duty period of the neutron generator must usually be restricted.

7.9 No significant radiation effects have been demonstrated in animals or humans at or below the maximum permissible occupational dose levels. By extrapolation from observations at much higher dose levels, it seems possible that long-continued irradiation, even at permissible levels, may have some deleterious effect. The probability of such effects is believed to be less than that of many other injuries due to physical and chemical health hazards to which we are daily exposed. Nevertheless, efforts *should* be made to avoid needless radiation exposure.

7.10 Individuals who are in no way associated with the operation of a source may be inadvertently exposed to detectable amounts of radiation. If such locations are outside the controlled area and accessible to the public, it may be difficult or impossible to regulate or predict the occupancy accurately. Because, in addition, a comparatively large number of individuals (including children) might be exposed, special safeguards must be provided so that exposure will be held below the non-occupational limits specified below.

7.11 The NCRP recommends a limit of 500 mrem per year for individuals in uncontrolled areas. This limit amounts to an average of 125 mrem per 3-month period and it may be desirable to administer the limit on this basis. Obviously, the surest way to attain this objective is to provide sufficient protection so that a dose of more than 10 mrem per calendar week cannot be received in the areas in question regardless of the period of occupancy; hence, this approach is strongly recommended. If higher dose equivalents can be received in such uncontrolled areas during a calendar week, it must be established that the prescribed limit for a year can not be exceeded or regular checks will be required to insure that individuals do not remain there long enough to exceed the above limits. These limits apply to all of the radiation from the several generators which contribute radiation to the same location. The place-

ment of new radiation sources in the vicinity may, therefore, require further reductions of the stray radiation escaping from existing ones.

7.12 The NCRP has also recommended limits for certain special groups. Report No. 32 (NCRP, 1966) specifies that persons under 18 years of age *shall not* be occupationally exposed to radiation. (They *should not* be employed or trained in an x-ray department, radioisotope laboratory, or industrial radiation facility.) Students under 18 years of age exposed during educational activities *should not* receive whole body exposure exceeding 0.1 rem per year due to their educational activity. To provide an additional factor of safety, it is recommended that each experiment be so planned that no individual receives more than 0.01 rem while carrying it out.

7.13 The 1957 recommendations of the NCRP (NCRP, 1957a) stipulate that "the maximum permissible dose to the gonads for the population of the United States as a whole for all sources of radiation, including medical and other man-made sources, and background, shall not exceed 14 million rems per million of population over the period from conception up to age 30, and one-third that amount in each decade thereafter. Averaging should be done for the population group in which cross-breeding may be expected". This limit has been set because of genetic effects rather than the likelihood of personal injury. Exposure within the limits given in Section 7.11 should meet the requirement.

7.14 It should be noted that unanticipated increases of the attainable output of neutron sources frequently become possible due to technical advances. Therefore, in the design of neutron protection it is advisable to make provision for later additional shielding. Alternatively, the original shielding may be somewhat overdesigned with this contingency in mind.

8. Gamma- and X-Ray Hazards Arising in the Operation of Neutron Sources

8.1 In practice, the presence of neutrons is almost invariably accompanied by x and gamma radiation. Whenever neutrons have energies greater than the lowest excitation level of nuclei in the environment of the source, inelastic scattering can occur in which the nucleus is left in an excited state. This process (having a cross section usually below 3 barns) leads to the emission of characteristic gamma radiation. Slowed-down neutrons are commonly captured which also leads to gamma radiation. Both of these processes must be anticipated at all neutron installations. A brief description of additional mechanisms of photon production in various types of neutron installations is given below.

8.2 *Low-Voltage Ion Accelerators (Below 400 keV).* Although used to produce neutrons free of gamma rays, such accelerators are usually strong sources of x rays. Some of these x rays may be produced by stopping of the bombarding particles in the target. However, the most important contribution is usually produced at the ion source by heavy negative ions and electrons released at the target or at the accelerating tube, which are back-accelerated to bombard the ion source and its supporting structures. This radiation may be reduced (but never entirely suppressed) by making all of the parts of the accelerator out of materials with low atomic number and by the application of positive potential to the target structure.

8.3 *High Voltage Ion Accelerators.* In addition to producing x rays, these accelerators become sources of gamma radiation because of the increased likelihood of nuclear excitation of the target material and the walls of the accelerator. The energies of the photon radiation are characteristic of the nucleus which is bombarded and may vary from a few tens of keV to about 20 MeV.

8.4 (α,n) *Sources.* Many (α,n) sources are also sources of intense gamma radiation because of concomitant emission in the radioactive chain (^{226}Ra, ^{223}Ra, ^{228}Th, ^{227}Ac, etc.). Even if the α-ray source is relatively free from primary gamma radiation (e.g., ^{210}Po, ^{239}Pu, ^{241}Am), a significant amount of high energy gamma radiation (e.g., 4.4 MeV

from ^{9}Be (α,n) ^{12}C) accompanies most reactions of interest (see Table 5, Appendix C).

8.5 *Photoneutron Sources.* (γ,n) emitters, such as radioactive sources and betatrons, are particularly hazardous sources of gamma radiation because the relatively small cross section of the reaction, as compared to the Compton and pair-production process, requires implicitly an overwhelming flux of photons for the production of a relatively small number of neutrons.

8.6 *Thermal Neutron Sources.* Such sources as nuclear reactors present the added hazard of gamma rays, which arise in fission and the capture of neutrons by nuclei. The cross sections for the capture of neutrons by nuclei vary widely (10^{-5} to 10^{6} barns), and the energy of the gamma radiation emitted varies from a few hundredths to about 10 MeV. Frequently the resulting nucleus is radioactive and also emits gamma radiation.

8.7 *Inelastic Scattering.* Fast neutrons, of energy greater than that of the lowest excitation level of the nucleus, can lose energy by leaving the nucleus in an excited state. This process (of cross section usually below 3 barns) results in the emission of gamma rays characteristic of the excited nucleus. This type of gamma radiation must be anticipated in all installations producing fast neutrons.

8.8 *Fission Sources.* Since the fission process always produces gamma rays and radioactive fission products, it is reasonable to expect a definite gamma hazard.

9. Measurement of Neutron Radiation

9.1 Neutrons are detected by observing the result of their interactions with atomic nuclei. Detection may be essentially instantaneous as in observing (a) the recoil nucleus from an elastic or inelastic scattering collision, (b) the charged particles arising from nuclear reactions, (c) the gamma ray emitted in inelastic scattering, (d) fission recoils or gamma rays accompanying nuclear fission, (e) the one or more gamma rays following neutron capture or a secondary process immediately folllowing the neutron capture. Detection at a later time may be achieved by observation of the radioactive disintegrations of either fission products or unstable nuclei produced by neutron capture. If the presence of neutrons has not been foreseen, and therefore no provision for immediate measurement has been made, assessments in terms of the degree of induced radioactivity may be very useful.

9.2 As in the case of photon radiation (x and gamma rays) the direct ionizing effects of neutrons are negligible and detection methods are based on ionization by charged particles produced directly or indirectly as a result of the processes described in Section 9.1. Hence neutron detectors commonly utilize devices that respond to other ionizing radiations and an important problem is discrimination between neutrons and other radiations, particularly gamma radiation. Examples of techniques used for such discrimination are pulse-height discrimination, pulse-shape discrimination, observation of recoil tracks in photographic emulsions, and paired ionization chambers. While discrimination may be relatively simple in a detector which needs only to give an indication of the presence of neutrons, the measurement of neutron flux density, or spectrum of absorbed dose, usually requires a detailed understanding of the operation of the detector in order to obtain acceptable accuracy. If radiation energies are less than several MeV, devices depending on delayed processes can usually be counted upon to discriminate against everything but neutrons, but as a rule, activation methods require a knowledge of the neutron spectrum for interpretation of the results. However, a series of detectors, such as neutron threshold detectors or resonance detectors, can furnish approximate information about the neutron spectrum. The problems of the dosimetry of mixed radia-

tions are discussed more completely in NCRP Report No. 23 (NCRP, 1960a), NCRP Report No. 25 (NCRP, 1961), and ICRU Report 10b (ICRU, 1964).

9.3 If the energy and direction of neutrons are known, it is possible to derive kerma from a measurement of fluence utilizing the data given in Appendix B. In neutron physics usage, fluence is commonly called *nvt*, and fluence rate or flux density, *nv*.

9.3.1 Measurement of thermal neutron flux density is usually carried out by observing the reaction rate of an element of known cross section, such as boron, lithium, manganese, or gold. For polyisotopic elements it may be necessary to know the isotopic composition of the sample for precise measurements. With a neutron detector whose cross section is proportional to $1/v$, where v is the neutron velocity, the reaction rate is a measure of the neutron density, n and is independent of the neutron velocity or energy spectrum. For N nuclei the reaction rate, R, in reactions per second, is

$$R = \int n(v) v \mathrm{N} \sigma(v)\, \mathrm{d}v$$

which for

$$\sigma(v) = \frac{\sigma_0 v_0}{v}$$

is

$$R = \mathrm{N}\sigma_0 v_0 \int n(v)\, \mathrm{d}v = \mathrm{N}\sigma_0 v_0 n$$

Here $n(v)$ and $\sigma(v)$ are the distribution in velocity of neutron density and cross sections. Because of this relation it is customary to tabulate for thermal neutrons only the cross section, σ_0, corresponding to $v_0 =$ 2200 meters/sec. Whereas n is the true neutron density, nv_0, the quantity obtained from a reaction rate measurement using a cross section, σ_0, is not the true neutron flux density and is called the "conventional flux density". The true neutron flux density (nv) for a Maxwellian distribution at temperature T is larger than the effective flux density (nv_0) by a factor $1.128\sqrt{T/T_0}$, where $T_0 = 293.59$°K (20.44°C). Thus measurements made with $1/v$ detectors using tabulated cross sections will underestimate slightly the true flux density (upon which the permissible flux densities are based). However this factor should be negligible in most radiation protection considerations.

9.3.2 Measurement of neutron flux density or fluence for intermediate neutrons is usually accomplished by observing neutron-induced

reactions such as $^{6}Li(n,\alpha)^{3}H$, $^{10}B(n,\alpha)^{7}Li$, or $^{3}He(n,p)^{3}H$; or by observing activation of resonance detectors such as In, Au, Co, Mn, Cu, and Na. Another technique is the use of a "black" detector—a detector so thick and of such large neutron absorption cross section that all neutrons are absorbed. Using materials such as ^{10}B and ^{6}Li, black detectors with flat response up to about 10 keV are possible. For further information on intermediate neutron measurements see NCRP Report No. 23 (NCRP, 1960a) and ICRU Report 10b (ICRU, 1964).

9.3.3 Methods of measurement of fast neutron flux density and fluence include: (a) Observation of proton recoils from accurately known hydrogen elastic scattering, H(n,n)H, in a counter telescope, nuclear emulsion, bulk organic scintillator, or other detector; (b) counting of nuclear reactions, such as $^{3}He(n,p)^{3}H$, or neutron-induced fission; (c) associated particle counting of neutron-producing reactions, such as $^{3}H(d,n)^{4}He$ or $^{2}H(d,n)^{3}He$, used with a positive-ion accelerator; and (d) moderation methods where the neutrons are slowed down in a moderator and detected as thermal neutrons. Method (a) requires knowledge of the incident neutron direction (except for the bulk organic scintillator), but can be used to obtain information about the neutron spectrum. Method (b) has the advantage of not requiring information about the neutron direction. Method (c) is capable of high accuracy, but can be employed only if the neutron source is an accelerator. Method (d) is sensitive to neutrons of all energies and can be made rather insensitive to neutron spectrum and therefore is suitable for flux density measurements for radiation protection. Calibration with a radioactive neutron source of known emission rate is usually required. Further information on fast neutron flux density and fluence measurements can be found in NCRP Report No. 23 (NCRP, 1960a), ICRU Report 10b (ICRU, 1964), and Marion and Fowler (1960).

9.4 Determinations of kerma and absorbed dose may be made using: (a) ionization methods as utilized in ionization chambers and proportional counters, (b) calorimetry, (c) chemical systems such as photographic film and liquid chemical dosimeters, (d) neutron fluence and spectrum measurements with the employment of fluence to kerma conversions such as are shown in Figure 3, Appendix B.I, or (e) special counting devices which have their response adjusted so that they are approximately proportional to kerma or absorbed dose over a range of neutron energies.

9.5 It seems appropriate here to indicate some general aspects of the use of radiation instruments rather than to discuss specific instruments.

9.5.1 It is advisable to utilize several different, independent monitoring methods rather than to place reliance on a single one. For ex-

ample, use of film badges without area monitoring might lead to a long delay in discovering a neutron hazard. By use of survey meters alone, without continuous monitoring, one might miss certain hazards altogether.

9.5.2 There *should* be methods (such as calibrations with a neutron source) to check periodically all neutron monitoring systems to insure that they are accurate. Neutron survey meters are notoriously prone to variations of sensitivity.

9.5.3 Instruments of unknown energy response which merely possess sensitivity to neutrons are of little use in radiation protection and *should* be used only to warn of the possible existence of a hazard.

9.5.4 Readings obtained with any instrument that depends upon the detection of individual particles must be suspected when the instrument is used for measurements in the vicinity of a pulsed neutron source. It will read correctly only if capable of time resolution of the particles received during the active part of the machine cycle. Otherwise the response of the instrument may not be proportional to intensity and may in extreme cases merely yield the pulsing rate of the generator.

9.5.5 Use of threshold detector foils or other continuously sensitive systems is strongly recommended at locations where a neutron radiation accident might occur (near a critical assembly, for example). Such a system *should* be integrated into the general monitoring program. Ease of recovery *should* be considered in the design of the system.

9.6 In addition to the NCRP reports concerned with the use of radiation instruments for neutrons, NCRP Report No. 23 (NCRP, 1960a) and NCRP Report No. 25 (NCRP, 1961), there is an extensive technical literature on this subject. The reader's attention is directed especially to reference books (Price, 1958; Hine and Brownell, 1956; Attix and Roesch, 1966, 1968; Attix and Tochilin, 1969; and Morgan and Turner, 1967) and the proceedings of several symposia (IAEA, 1961; IAEA, 1963; IAEA, 1967; ENEA, 1963; and USAEC, 1966).

III. RADIATION PROTECTION IN INSTALLATION AND OPERATION OF NEUTRON SOURCES

10. Types of Neutron Sources

10.1 *Radioactive Neutron Sources.* Radioactive neutron sources are of two types, (α,n) and (γ,n). In the (α,n) source, the alpha emitter is mixed with the target material or may even be alloyed with it. In (γ,n) sources the gamma-ray emitter is usually surrounded by the material (e.g., Be or D_2O) in which the neutron-producing reaction takes place.

10.1.1 Polonium-210, plutonium-239, and americium-241 in combination with beryllium or other target materials are suitable radioactive alpha-neutron sources from the standpoint of large neutron yield and low gamma-ray yield. These elements, however, are among the most dangerous ones when ingested or inhaled. Special precautions must be taken to prevent their escape by providing durable, sealed containers. Wipe tests to check for leakage of alpha-ray emitting material *should* be made frequently to verify that the source is not leaking.

10.1.2. Radium-226, in equilibrium with its daughter products, is another alpha emitter of high toxicity often used in neutron sources. Hermetic containment of radium-226 is exceptionally important because of dangers due to its daughter product, radon. Radium sources and all photoneutron sources produce high gamma-ray intensities which constitute a serious hazard which, for an unshielded source, usually predominates over the neutron hazard.

10.1.3 The neutron energy spectra from most (α,n) sources extend up to the neighborhood of 10 MeV while neutrons from radioactive (γ,n) sources are generally below about 1 MeV. Tables of the characteristics of (α,n) and (γ,n) sources may be found in ICRU Report 10b (ICRU, 1964) and in an article by Amaldi (1959).

10.2 *Spontaneous Fission Neutron Sources.* For a number of heavy even-even nuclei, spontaneous fission is energetically possible. These sources are attractive because of their fission-like spectrum, relatively low gamma-ray yield, and often, also because of their small mass. The spontaneous fission source materials usually produce many alpha rays per neutron and precautions against inhalation and ingestion must be taken. The relatively well-known spectrum would make these sources

suitable as dosimetry standards, particularly for instruments employed at nuclear reactors. Tables giving their characteristics may be found in NCRP Report No. 23 (NCRP, 1960a) and ICRU Report 10b, (ICRU, 1964).

A nuclide of especial interest is californium-252. It is becoming available in appreciable quantities and may, in its encapsulated form, find use in both research and radiotherapy.

10.3 Constant-voltage accelerators, as used in the production of neutrons, produce a high voltage by conveying charge mechanically with a belt (Van de Graaff accelerator) or by some system of producing an alternating voltage which can be applied to each of a series of rectifiers (Cockroft-Walton, transformer-rectifier, etc.). Particle energies of about 20 MeV are obtainable on some types of tandem Van de Graaff accelerators for protons and deuterons, and still higher energies for alpha particles and heavy ions. Small accelerators using deuterons of one or two hundred keV energy can produce a large number of 14-MeV neutrons. As beam energy increases, more and more nuclear reactions become available. An accelerator capable of imparting all energies up to 10 MeV to both protons and deuterons can produce monoenergetic neutrons at all energies up to 27 MeV. For this reason, constant-voltage accelerators are useful in the calibration of neutron radiation instruments. Large numbers of neutrons ($>10^{12}$/s) can be produced by bombarding a thick beryllium target with deuterons or alpha particles from a constant-voltage accelerator. Further information, particularly on monoenergetic neutrons from accelerators, may be found in Marion and Fowler (1960). Data on neutron production by nuclear reactions is given in Appendix C.

10.4 High-frequency positive ion accelerators include the cyclotron, the synchrocyclotron, the proton synchrotron, and the heavy ion linear accelerator. As the beam energy increases, neutron production generally increases, and the neutron spectrum spreads over a wide range of energies.

Protons above 10 MeV will produce neutrons when striking almost any material. The beam from all of these accelerators is always pulsed. Many neutrons often come from the accelerator itself as well as from the intended target. The large number of neutrons associated with these machines entails problems of induced radioactivity in the materials of the accelerator, in dust and in the air of the accelerator and experiment rooms. Thus both external and internal radiation hazards can be present.

10.5 High-frequency electron accelerators include circular accelerators such as the betatron and synchrotron and linear accelerators. The

primary radiation hazard is x rays produced by accelerated electrons striking the target or other material. However, these accelerators can produce large numbers of neutrons from a target such as tungsten due to the (γ,n) reaction occurring in the target. With the use of a uranium target a still higher yield of neutrons is obtained since the high energy photons produced in the target produce both photo-neutrons by the (γ,n) reaction and neutrons by photo-fission (γ,f). Most of the neutrons are produced with energies of a few MeV, while fewer neutrons are produced with energies up to near the maximum bremsstrahlung energy of the accelerator. These few neutrons of initially high energy often determine the shielding requirement of electron accelerators. (See NCRP Report No. 31, NCRP, 1964).

10.6 *Nuclear Reactors.* Neutron production in reactors occurs as a result of the fission process, which is maintained by means of a carefully controlled chain reaction. In the usual operating mode the reactor is critical and the number of fissions occurring is substantially constant in time. This is achieved by a definite arrangement of fuel (uranium or plutonium), moderator (light water, heavy water, beryllium or graphite), and adjustment of neutron absorbers (control rods). In a nuclear reactor neutrons are produced over a wide range of energies from thermal to over 15 MeV, the number of neutrons at energies above 10 MeV being relatively small. Many gamma rays are also present from the fission process itself, from radioactive fission products, and from neutron-induced reactions. The most probable neutron energy in thermal reactors is in the thermal region, whereas in "fast reactors" it may be a few hundred keV. Many approximate measurements of reactor neutron spectra are available (NCRP, 1960a; IAEA, 1963).

11. Shields

11.1 Adequate shielding against neutrons will often attenuate gamma radiation to acceptable levels at both reactors and accelerators. Water and other hydrogenous shields may constitute an important exception to this rule. In the use of radioactive sources, and in particular photoneutron sources, gamma shielding is a separate and often more important problem.

11.2 In the design of buildings planned to contain neutron generators, shielding *should* be a prime consideration because of size, weight, and cost.

11.3 Ordinary or heavy aggregate concrete or earth are the recommended materials in most installations. Any economy achieved by the use of water-filled tanks is likely to be offset by maintenance difficulties. In addition, evaporation can lead to dangerous loss of shielding effectiveness, although evaporation may be retarded by the addition of oil.

11.4 Both paraffin and oil are fire hazards, and neither should be used in large stationary shields.

11.5 It is often necessary to operate with temporary shielding, in which the shield is not cast into place but rather is built up of separate blocks. As all such installations are subject to flaws or cracks which are left in the assembly, it is particularly important that a detailed survey be made prior to routine operation of the source. In general it is found that even a carefully laid unmortared concrete block shield is less effective than a monolithic poured structure. Lead bricks similarly show reduced attenuation. All vertical cracks *should* be staggered to reduce leakage. Gravity is usually sufficient to minimize horizontal cracks.

11.6 Even though shields may be satisfactory when installed, they may deteriorate, either suddenly or gradually, so that it is necessary to monitor the radiation outside routinely. Examples of such deteriorations are the loss of water from a shield tank or from a hydrogenous shield material, the development of cracks in concrete due to settling, or the loss of hydrogen due to radiation damage in paraffin or oil. Reactors in water pools should be equipped with proper monitor systems to warn of lowered water level.

11.7 Ducts that penetrate shields will provide a path for radiation and may cause objectionable levels in otherwise adequately shielded areas. This effect may be minimized by proper placement of the ducts or by providing bends or offsets. Structural elements such as steel beams may act as ducts by providing low attenuation pathways through a shield for neutrons in certain energy ranges.

11.8 Methods of shielding calculations are outlined in Appendix E.

12. Signs and Barriers

12.1 The purpose of signs and barriers is to alert workers or the public to possible radiation hazards and thus to reduce unnecessary exposures. These aims *should* be accomplished with the minimum interference with the work being done. The types of signs and barriers *should* be graded according to the magnitude of the hazard involved and will depend somewhat on the nature of the work, the degree of supervision, the type of personnel, the location of the area in question and other factors. Thus, signs and barriers are often provided as a warning rather than to indicate exclusion and activities may frequently be carried out within the associated areas if adequate exposure controls are provided. To ensure the credibility of signs it is essential that they be removed when no longer applicable.

12.2 Areas with very high exposure rates *shall* be locked or otherwise be made effectively inaccessible. 100 mrem in any one-hour period is a typical level above which this type of control *should* be provided.

12.3 For lower but still substantial exposure rates, where access must be controlled to avoid overexposure, areas *should* be clearly delineated by the use of appropriate signs and barriers. 100 mrem possible exposure during a 40-hour week is a typical level above which this *should* be done in work areas. Signs alone may suffice under some circumstances but, for the higher levels, ropes, fences or even barricades with controlled check-in and check-out *should* be utilized.

12.4 When exposures may be in the 10–100 mrem per week range, a controlled area *should* be established with suitable labelling at the entrances, but signs and barriers will not normally be needed within the area.

13. Procedures in Case of Significant Overexposure

13.1 According to the principles discussed at length in NCRP Report No. 17 (NCRP, 1954), the radiation exposure status of an individual is altered if, once in his lifetime and within a period of one month or less, he is exposed to a dose equivalent exceeding 25 rems to the whole body or a major part thereof. In the following, a "significant overexposure" will be considered to be one in which an individual receives in one month a dose equivalent that exceeds a level that *shall* be established by responsible management. The level set depends on circumstances but *should not* be less than 5 rems and *shall not* be more than 25 rems. Significant overexposures must be referred to recognized experts in radiological physics and medical radiology for joint consideration and appraisal.

13.2 Since the QF applies only near permissible levels, dose equivalents beyond about 25 rems become meaningless and the absorbed dose (in rads) *should* be specified together with appropriate information on radiation quality. In any significant overexposure, every effort *should* be made to evaluate such large absorbed doses as accurately and as soon as possible. The extent to which any clinical manifestations may contribute to an understanding of human radiosensitivity is directly related to the availability of a reliable dose estimate.

13.3 The scope of the investigative program to determine the dose will depend on the seriousness of the exposure and the amount and type of data available from radiation monitoring devices at the scene of the accident. Therefore, no hard and fast rules can be given to govern the course of action to be taken. However, experience reported in connection with the dosimetry of past nuclear accidents has made it possible to outline some general procedures which may be followed, together with the dosimetric principles involved. A detailed discussion of the principal considerations of accident dosimetry is given in Appendix F. A list of emergency measures which are prerequisite to dose assessment is given in Sections 13.4 and 13.5. However, it must be stressed that there may be other steps of crucial importance which must be carried out promptly after an accident. Thus steps must be taken immediately to ensure that there is no recurrence (e.g. of criticality). Since these other safety con-

siderations are not directly related to accident dosimetry, they are not discussed here.

13.4 In the management of individuals who may have received a substantial accidental overexposure, the following steps are indicated. The order and degree of execution will vary depending on local circumstances.

13.4.1 Escort evacuees from the scene of accident to a convenient area in which radiation levels are low enough to permit contamination and activation measurements. Inquire if any of them saw a flash of light at the time of the accident.

13.4.2 Identify persons with external contamination and initiate appropriate decontamination procedures.

13.4.3 Check contamination-free persons for neutron exposure by gamma-ray measurement of body mid-section or by indium foil monitor. Record readings.

13.4.4 Interview exposed persons to ascertain position, orientation, and movement relative to the radiation source. Complete retention of all information is facilitated by use of a tape recorder. Record information about positions, orientation and movement on a plan drawing of the building or the area in which the accident occurred. Try to determine the time of the accident and obtain photographs of the scene of the accident.

13.4.5 Collect all monitoring devices worn by exposed persons. Record position of the device on the body.

13.4.6 Obtain blood samples from exposed persons prior to any oral or intravenous administration of inactive sodium. Secure hair samples, recording the exact location (e.g., position on head or body) from which hair is taken. Samples should be preserved by freezing.

13.4.7 Collect personal items likely to contain induced radioactivity, e.g., pens and pencils, coins (Cu, Ni, Ag), keys, match folders (Cl in match heads, P on striking surface) jewelry, metal buttons, buckles, garters, eyeglasses (Si, Na, Ba in lenses), dental appliances, etc. Record position on body.

13.4.8 Forward well-labeled blood and hair samples, monitoring devices, and personal items to a radioactivity laboratory, together with all recorded data and information.

13.4.9 If facilities are available, and with the consent of the attending physician, arrange for whole-body counting of exposed persons.

13.5 The following procedures *should* be carried out at the accident scene.

13.5.1 As soon as it is safe to do so, recover monitors and dosimeters installed in the vicinity of the accident.

13.5.2 Search for small objects which may possess induced radioactivity, e.g., tools, components of experimental apparatus, scrap materials, solder (Sn), semi-conductor devices (Ge, Si), dry batteries (Mn, Zn), thermometers (Hg), etc. Without disturbing the existing arrangement of furniture and equipment, collect such objects in sufficient number to supplement the installed monitors in their function of defining the neutron field. Label each item and record its location (and elevation) on a plan drawing of the area.

13.5.3 Forward monitoring devices and miscellaneous objects to the dosimetry laboratory, together with pertinent recorded information and an indication of the contamination status of each item. Use protective coverings to prevent spread of contamination.

13.5.4 Check all operating radiation detection equipment in the area for recorder charts which may indicate the time and/or intensity of a radiation burst. Note both the time at which a chart record is terminated and the identification numbers of the recorder (so that chart speed may be established). Send labeled charts to the dosimetry laboratory.

13.5.5 If the accident involved a criticality, arrange for radiochemical analysis of reaction material to determine total number of fissions.

13.5.6 Interview persons familiar with operations related to the accident to obtain additional information of possible value in evaluating exposure, e.g., normal position and movement-sequence of an equipment operator.

IV. RULES FOR PROTECTION AGAINST NEUTRON RADIATION

14. Scope of Rules

14.1 The rules set forth below are considered essential for the avoidance of hazards attending exposure to neutron radiation. Other radiations are considered only insofar as they might occur simultaneously with neutrons and add to the exposures incurred. Other reports issued by the NCRP deal more explicitly with protection against other ionizing radiations. The present rules are not concerned with any electrical, mechanical, toxicological, and other non-radiation hazards that might arise in the operation of neutron sources, except as they affect radiation safety.

14.2 A further restriction applies to the case of fissile material. The rules set forth below extend to protection during normal operation at power levels anticipated. The prevention of abnormal conditions that entail particularly severe radiation hazards is a complex technological problem that will not be discussed here.

15. Rules Pertaining to Maximum Permissible Dose Equivalent

15.1 *Occupational Exposure.*

15.1.1 The maximum permissible dose equivalent (MPD) to the most critical organs (whole body, head and trunk, active blood-forming organs, gonads and eyes) accumulated at any age, *shall not* exceed 5 rems multiplied by the number of years beyond age 18, and the dose equivalent in any 13 consecutive weeks *shall not* exceed 3 rems.

15.1.2 The dose equivalent to the skin of the whole body *shall not* exceed 30 rems per year or 10 rems in any 13 consecutive weeks.

15.1.3 The dose equivalent to the hands and forearms, feet and ankles, *shall not* exceed 75 rems per year or 25 rems in any 13 consecutive weeks.

15.1.4 Exposure of pregnant and potentially pregnant women is limited by the fact that the maximum permissible dose equivalent to the fetus *should not* exceed 0.5 rem.

15.2 If detailed information on the nature of the ionizing radiation is not available, the $\overline{\text{QF}}$ *shall* be assumed to be 10. If the portion of the tissue dose contributed by the various radiations is known, each absorbed dose in rads *shall* be multiplied by the appropriate $\overline{\text{QF}}$ to obtain the dose equivalent in rems. The $\overline{\text{QF}}$ of electrons (whether primary or produced by electromagnetic radiation) *shall* be taken as 1.0. The $\overline{\text{QF}}$ of neutrons *shall* be taken as 10, except that, if the distribution of neutron energies is known, the $\overline{\text{QF}}$ values in Table 2 (Section 7) may be applied. If the dose equivalent is determined by measurements of $D(L)$, the dose distribution in LET, the integral in equation 1 (see Section 2.2) *shall* be evaluated.

15.3 Radiation levels outside controlled areas, due to sources within controlled areas, *shall* be such that no person *shall* incur a dose equivalent of more than 500 mrem per year. It *shall* be the duty of the radiation protection officer to assure himself that there is no likelihood that any person while remaining in these areas will exceed this limit.

15.4 The dose equivalents received by students *shall not* exceed the limits stipulated in Section 7.12.

16. Radiation Protection Officer

16.1 Personnel responsible for work with neutron sources *shall* also be responsible for radiation safety. If a neutron source is capable of delivering more than 100 mrem per work week (due to *all* ionizing radiations) in accessible regions inside or outside of any externally applied shielding, a radiation protection officer *shall* be designated by the management concerned. His responsibilities *shall* include:

16.1.1 Provision of technical assistance in the planning and execution of work insofar as radiation safety considerations are involved.

16.1.2 Appraisal of operation of the source with regard to the radiation safety rules set forth in this report.

16.1.3 Notification of personnel working near the source of any special hazards that may exist.

16.1.4 Awareness of exposure of such personnel from additional sources of ionizing radiation.

16.1.5 Reporting of radiation hazards or unsafe practices to the proper authorities for suitable action whenever necessary.

16.2 The radiation protection officer *should* be familiar with the contents of this report, and *shall* have sufficient training and experience to understand and apply pertinent provisions. A user of the source or a person employed in other capacities may qualify as radiation protection officer. A radiation protection officer may delegate duties but not ultimate responsibility. He *shall* be guided by advice from qualified experts, if necessary.

16.3 The radiation protection officer *shall* keep records of personnel exposure and area dose levels.

16.4 The radiation protection officer *shall* be informed of any changes in the mode of operation of the source.

17. Radioactive Sources

17.1 Neutron sources containing materials which, if inhaled or ingested, would constitute a potential radiation hazard *shall* be sealed securely or handled under conditions that otherwise minimize the hazards involved.

17.2 A neutron source having a surface dose equivalent exceeding 10 mrem per calendar week (due to all ionizing radiations emitted) *shall* be marked with a label or stored in a labeled container. The label *shall* contain information on the nature and strength of the source.

17.3 Any neutron source having a surface dose equivalent rate of more than 200 mrem/h, or a dose equivalent rate of more than 10 mrem/h at 1 meter, *should* be stored in a labeled container conforming with the requirements set out in Section 17.4

17.4 When the source is in its storage container, less than 200 mrem/h *shall* be delivered at any container surface and less than 10 mrem/h at 1 meter from the container.[4] These requirements need not be fulfilled if the regions around the source are marked as described in Section 17.5

17.5 When such a source is removed from its storage container, appropriate signs and barriers *shall* be provided in accordance with the principles stated above in Section 12.

17.6 Wipe tests of radioactive sources *shall* be carried out at least semi-annually.

[4] These values have been chosen so that containers may be used for purposes of source shipment in accordance with regulations of the U.S. Department of Transportation.

18. Accelerators

18.1 Appropriate signs and barriers *shall* be provided for accessible locations where appreciable radiation exposures may be received, in accordance with the principles stated in Section 12.

18.2 If the installation is provided with a shield segregating accessible locations, and if a dose rate in excess of 10 mrem/h can be received inside the shield, provisions *shall* be made for audible indication that the beam is about to be turned on. The indication may be either intermittent or continuous but must last for at least a 10-second period prior to beam production.

18.3 If exit from the shield cannot be effected without motion of doors or other similar impediments, provisions *shall* be made that:

18.3.1 Such doors are interlocked with the accelerator controls in such a way that neutron production is impossible with the door open.

18.3.2 The doors can be opened from the inside of the enclosure.

18.3.3 At least one clearly marked crash button is located within the enclosure that, when pressed, will suspend accelerator operation in such a way as to make neutron production impossible. It *shall* be possible to readily reach one such button from any point inside the enclosure within 5 seconds of onset of the audible warning signal.

18.4 At any installation where dose rates in excess of 100 mrem per work week may be received outside the shielding, it *shall* be the duty of the radiation protection officer to check regularly to insure that persons do not receive doses in excess of the permissible levels (see Sections 15.1–15.2). "Persons" includes operators, experimenters, visitors and individuals employed in maintenance or other duties not directly associated with machine operation. Such checks *shall* include a daily assessment of the operations of the accelerator.

19. Reactors

19.1 On starting up a reactor for the first time, every accessible region *should* be surveyed for radiation, and appropriate adjustments *should* be made in the shielding.

19.2 Because it is possible for radiation leaks to develop, surveys *should* be made at least annually and whenever changes have been made in the shield or its perforations.

19.3 Because reactor radiation levels increase strongly on startup, an audible warning system *shall* be installed to insure that all personnel in the vicinity of the reactor are made aware that startup is planned. Personnel *shall* be able to communicate with the reactor operator within a period of time that is less than the warning period.

19.4 In the vicinity of the reactor, continuous visual indication *shall* be provided to inform personnel whether the reactor is in operation.

19.5 Rigorous procedures *shall* govern the changes of shielding. These *shall* include consultation with the radiation protection officer.

19.6 Upon first removal of a shielding block, or first testing of a beam trap placed behind a movable shutter, monitoring *shall* be performed to assess the existing radiation hazard.

19.7 Appropriate signs and barriers *shall* be provided for accessible locations, where appreciable radiation exposures may be received, in accordance with the principles stated in Section 12.

19.8 In the event of an accident it may be necessary to evacuate the building quickly, and probably also the area outside. Therefore, a plan *shall* be devised for this evacuation. Responsible persons *shall* be designated and available at all times for its execution. All personnel normally in the environs *shall* be made aware of the plan. Practice evacuations *should* be performed.

20. Surveys at Accelerators and Reactors

20.1 During tune-up and initial operation, surveys of both the gamma- and neutron-radiation dose equivalent rate at accessible locations outside the shielding *shall* be performed as soon as doses in excess of 2.5 mrem/h are likely to be delivered.

20.2 Prior to routine operation, every accelerator or reactor *shall* be surveyed. Such surveys *shall* be repeated whenever operating conditions are changed in such a way that the neutron- or gamma-hazard may change significantly. In the absence of any such changes, surveys *shall* be made at least once a year. In installations where liquid shields are employed, a survey *shall* be performed at least once every 6 months.

20.3 At installations where liquid shields are employed, special precautions *shall* be taken to insure that the liquid is maintained at the desired level.

20.4 Individual monitoring equipment *shall* be provided and utilized when individuals are likely to receive exposures of more than 25 percent of the limits specified in Section 15.1.

20.5 When individuals will be, or are likely to be, exposed to dose equivalent rates in excess of 30 mrem/h, instantly-indicating monitoring equipment adequate for effective exposure control *shall* be in continuous operation.

21. Health

21.1 Prior to employment, each person working more than occasionally in areas where significant neutron doses may be received *shall* have a medical examination. The examination *should* be directed toward determining the normal or presumed "preirradiation" condition of the worker, and toward taking special precautions if any abnormalities exist that might later be confused with radiation damage. The examination *shall* include a complete blood count, with determination of erythrocyte, leukocyte, and platelet levels, and differential W.B.C. An eye examination *shall* include a determination of vision with and without glasses, dilation of pupils and examination of the lenses with the slit lamp (corneal microscope). A history of past exposure (occupational or radiotherapy) *should* be provided.

21.2 Blood counts *shall not* be used as a method of personnel monitoring.

22. Overexposure

22.1 Any exposure in excess of the permissible limit *shall* receive the immediate attention of the radiation protection officer and the management. Corrective measures *shall* be instituted to prevent recurrence and, if necesary, steps *shall* be taken to minimize further exposure so that the individual's accumulated dose equivalent returns to the level permitted for his age.

22.2 In the case of significant overexposure, as defined in Section 13.1, a considerable effort *shall* be made to determine the dose received, the portion of the body exposed, and the character of the radiation. A competent panel, including a physician familiar with radiation risk and injury, a physicist, the radiation protection officer, and a responsible administrator, *shall* review the circumstances of the exposure, and any medical findings that might be present at higher doses, and decide on the advisability of further investigations. They *shall* investigate the reasons for the overexposure and undertake all reasonable efforts to prevent repetition of overexposures.

APPENDIX A

Definitions

The following definitions are given for purposes of clarification of the contents of this report. In some instances they may differ somewhat from common usage. Many of the quantities and units defined below have been the subject of extensive analysis by the International Commission on Radiation Units and Measurements (ICRU). This information has been published as ICRU Report 11 (ICRU, 1968) where precise definitions may be found.

absorbed dose: The absorbed dose of any ionizing radiation is the energy imparted to matter by directly or indirectly ionizing radiation per unit mass of irradiated material. The special unit of absorbed dose is the rad.

accessible location: Any region around a source of ionizing radiation that can be reached without rupture of structures or without use of specially designed tools not generally available.

barn: A unit of area used in expressing a nuclear cross section. 1 barn = 10^{-24} cm^2.

capture: A process in which a neutron becomes part of the nucleus with which it collides without the release of another heavy particle.

controlled area: A defined area in which the exposure of persons to radiation or to radioactive material is under the supervision of a radiation protection (safety) officer.

cross section: Effective target area for a specified nuclear interaction. The cross section is a measure of the probability for the interaction. It is usually expressed in barns.

directly ionizing radiation: Radiation composed of electrically charged particles that are capable of ionizing by collision.

dose equivalent: The product of absorbed dose and the quality factor (QF). (Other modifying factors may also be employed.) The special unit of dose equivalent is the rem; the dose equivalent is in rems if the absorbed dose is in rads. In the absence of recommended simple terminology, values of dose equivalent or dose equivalent rate will be said to "exist at" or be "delivered to" some location; the meaning being that these values would be the most significant ones received by an individual if he were at that location. This quantity should be used for protection purposes only.

elastic scattering: Scattering collision in which the sum of the kinetic energies of neutron and target nucleus remains unchanged after the collision.

electron volt (eV): A unit of energy equal to the kinetic energy gained in a vacuum by a particle having one electronic charge when it passes through a potential difference of 1 volt; 1 eV = 1.60×10^{-12} erg.

energy fluence: Time integral of energy flux density.

energy flux density: Radiation energy which, per unit time, enters a sphere per unit of cross sectional area of that sphere.

exposure: The exposure of x or gamma rays is the total charge produced due to ionization in air by electrons that are ejected per unit mass of air. The special unit of exposure is the roentgen.

fast neutrons: Neutrons of energies above about 10 keV.

fluence (particle fluence): Time integral of flux density.

flux density[5] (fluence rate): The number of particles which, per unit time, enter a sphere per unit of cross sectional area of that sphere. In the case of neutrons it is usually expressed in $cm^{-2}s^{-1}$.

heavy particle: Any particle having a rest mass greater than that of the electron.

indirectly ionizing radiation: Radiation composed of uncharged particles which are capable of releasing charged particles when interacting with matter.

inelastic scattering: Scattering collision of neutron with attendant loss of kinetic energy which is expended in excitation of the target nucleus.

intermediate neutrons: Neutrons of energies of about 1 eV to 10 keV.

kerma[6]: The total kinetic energy of directly ionizing particles ejected by the action of indirectly ionizing radiation per unit mass of specified material.

kilo electron volt (keV): 1,000 eV.

kilovolt (kV): A unit of electrical potential equal to 1,000 volts. The term is also used to characterize the radiation emitted by x-ray tubes operating at a given potential.

linear energy transfer (LET): The average energy lost by a directly ionizing particle per unit distance of its travel in a medium. As here employed the term is to denote LET_∞, i.e., the entire energy loss of charged particles per unit distance with inclusion of all energy carried by charged particles (e.g., delta rays) which are not considered separately.

moderator: Material used to moderate, i.e., slow down, neutrons. Neutrons lose energy by scattering collisions with nuclei of the moderator.

non-elastic interaction: Any interaction other than elastic scattering.

nuclear reaction: Interaction between two or more particles including at least one nucleus and leading to emission of different particles.

qualified expert: A person suited by training and experience to perform appropriate radiation surveys, to oversee radiation monitoring, to estimate the degree of radiation hazard, and to advise regarding radiation hazards.

quality factor (QF): A factor which is used in radiation protection to weight the absorbed dose with regard to its presumed biological effectiveness insofar as it depends on the LET of the charged particles. The quality factor is a function of the LET of the charged particles that deliver the absorbed dose. The charged particles traversing irradiated matter usually have a range of values of LET and the term is then the average, or mean, quality factor, $\overline{QF}$, which is obtained by a weighting procedure given in Section 5.1.

rad: Special unit of absorbed dose. 1 rad is equal to 10^{-2} joules kg^{-1} or 100 ergs g^{-1}.

radiation protection (safety) officer: The person directly responsible for radiation protection.

radioactive neutron source: A neutron source consisting of a combination of

[5] The term "flux" has frequently been employed for this quantity but this name is deprecated.

[6] In this report the specified material precedes the word "kerma", e.g., where the specified material is tissue, the expression "tissue kerma" is used.

radioactive material and suitable target material. Neutron production occurs as a result of (α,n) or (γ,n) reactions.

relative biological effectiveness (RBE): Biological potency of one radiation as compared with another. The RBE of radiation A with respect to the reference radiation, B, is defined in terms of the absorbed doses, D_A, and, D_B, which produce equal biological effect. It is equal, therefore, to D_B/D_A. The standard of reference used in this report is moderately filtered 200 kVp x radiation, which thus has an RBE of 1. The use of this term is to be restricted to radiobiology and it should be distinguished from the quality factor which is employed in radiation protection. (See footnote 2, page 2.)

rem: Special unit of dose equivalent.

shall: Necessary to meet currently accepted standards of protection.

should: Indicates advisory requirements that are to be applied when practicable.

special unit: Unit reserved for one quantity only. Other units may also be employed for the same quantity. Thus the rad is the special unit of absorbed dose which may also be expressed in ergs/g, J/kg etc.

thermal neutrons: Strictly, neutrons in thermal equilibrium with their surroundings. In this report, all neutrons with energies of less than about 1 eV are termed "thermal".

week, calendar: 7 consecutive days.

week, work: A combination of time intervals adding up to 40 hours within 7 consecutive days.

APPENDIX B

Depth Dose

Radiation protection is based on the dose equivalent received by critical organs. Since these, in general, may be located at all depths, the principal objective must be an evaluation of the maximum dose equivalent in irradiated persons. When the human body is exposed to neutrons, extensive attenuation and scattering take place and consequently this quantity is not easy to evaluate. There are very few applicable experimental determinations available at present and certain protection recommendations must be based on calculations.

NCRP Report No. 20 (NCRP, 1957b) contains the results of such calculations for neutrons of energies ranging from thermal to 10 MeV which are perpendicularly incident to a 30-cm slab of tissue. These results were obtained by a Monte Carlo calculation with all major interactions with all major elements considered and utilizing experimentally determined interaction cross sections. In the present report this work has been refined by the replacement of the slab by a circular right cylinder. In some instances this reduction of the scattering mass has resulted in somewhat higher permissible fluence values.

A less precise but sufficiently accurate approximation may be obtained on the basis of general statistical models of the intranuclear cascade and nuclear evaporation (see Section 6.1.5.). The principal advantage of this approach is the possibility of extending the calculation to many of the much higher neutron energies which are of importance at some of the larger accelerators.

The two methods overlap in the energy range between 0.5 and 14 MeV and the agreement is satisfactory. The maximum dose equivalent per unit neutron fluence was used to derive the maximum permissible flux densities given in Table 2 (Section 7). Up to 14 MeV a cylindrical model was chosen and cross sections based on experiment were used in the calculations. Above this energy, a slab geometry and calculated cross sections were employed.

B.I. Dose Distribution in a Cylindrical Phantom for Neutron Energies Up to 14 MeV[7]

The depth-dose data reported here for neutrons of energies not exceeding 14 MeV were obtained by using a Monte Carlo-type code which is essentially the same as that described by Auxier et al. (1968) and consequently only a summary discussion is given with emphasis on the changes made for these calculations.

The phantom used is a right circular cylinder of height 60 cm and radius 15 cm, and the density is taken as 1; thus the total mass is about 42.4 kg. Only the four principal elements present in tissue are considered, the composition being homogeneous and H, C, N and O being respectively present in amounts of 6.169, 1.258, 0.107, and 2.333, all $\times 10^{22}$ atoms/g. The results reported here are for monoenergetic broad beams of neutrons with the beam direction parallel to the base of the cylinder. Results are reported for neutron energies of 14, 10, 7, 5, 2.5, and 1 MeV; 500, 100, 10, and 1 kev; and 100, 10, 1, and 0.025 ev.

The subregions over which dose (D) and dose equivalent (DE) have been averaged are displayed schematically in Figure 2. A Cartesian system of coordinates is used with origin at the center of the base of the cylinder and with the x-axis parallel to the direction of the neutron beam but with the neutrons initially moving in the negative x-direction. Then the subregions all lie between the planes $y = -2.5$ cm and $y = +2.5$ cm or between the planes $x = -2.5$ cm and $x = +2.5$ cm. The volume elements were selected primarily to provide estimates of D and DE along two lines parallel to the base of the cylinder and intersecting at the center of the cylinder, one line being parallel to the direction of the neutron beam and the other perpendicular to the beam direction. These two lines will be termed Traverse 1 and Traverse 2, respectively. They are displayed in Figure 2 and defined mathematically by Eq. (BI-1) below.

Traverse 1	Traverse 2	
$y = 0$	$x = 0$	(B1-I)
$z = 30$	$z = 30$	
$-15 \leq x \leq 15$	$-15 \leq y \leq 15$	

The volume elements used for averaging are parallelpipeds except where they are bounded by the lateral surface of the cylinder. The dimensions are 2 cm x 5 cm by 30 cm in the x, y, z directions, respectively, for the

[7] The NCRP is greatly indebted to W. S. Snyder for this contribution.

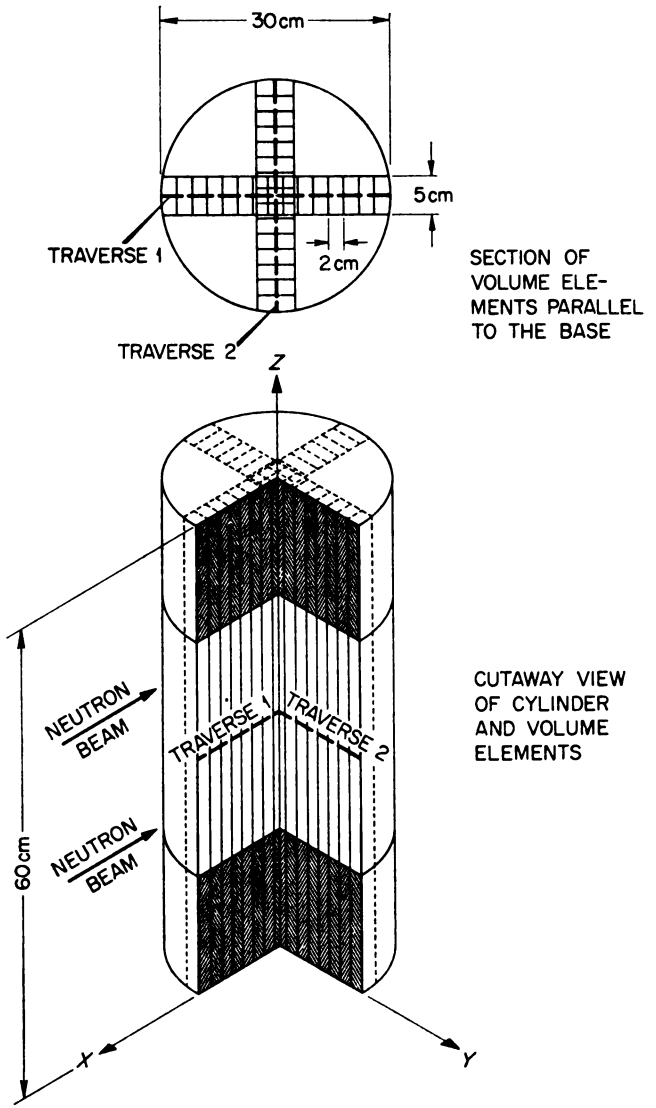

Fig. 2. Cylindrical phantom and volume elements for averaging of dose and dose equivalent.

volume elements centered on Traverse 1 and 5 cm x 2 cm x 30 cm in the x, y, z directions, respectively, for those volume elements centered on Traverse 2. Averaging over a volume element with such a large dimension as 30 cm requires some justification. The calculations reported in Auxier et al. (1968) had demonstrated that D and DE varied only

slowly with z, i.e., with displacement parallel to the axis of the cylinder except near the top and bottom. In fact, the differences found when dose was averaged separately in 5 equal layers parallel to the base were barely significant for the three central layers as compared with the statistical variation of the results. Hence, an averaging of dose over a considerable distance in the direction of z was tolerated, and the volume elements centered on Traverses 1 and 2 were taken to extend over the range $+15 \leqslant z \leqslant 45$. As a further check, these quantities were also averaged over volume elements with the same x and y dimensions but extended over ranges $0 \leqslant z \leqslant 15$ and $45 \leqslant z \leqslant 60$ cm and confirmed that the variations of D and DE with z were within the ranges indicated in the paper by Auxier et al. (1968). These volume elements adjacent to the bases of the cylinder are shaded in Figure 2. By symmetry, D should be identical at corresponding points in these peripheral volume elements, and hence the estimates were averaged for these pairs of volume elements. The same applies for DE.

Thirty-three different interactions were considered, and these are listed in Table 3. The first four are the elastic collisions in which energy and momentum of the struck nucleus and neutron are conserved. Anisotropy of scattering was taken into account only by the use of the P_1 approximation (Auxier et al., 1968). The interactions 5–16 are inelastic with one or more photons being produced. The gamma rays in reactions 5–7 and in 13–16 correspond to excitations of different energy levels in the nucleus. In the case of O (reactions 13–16) only two levels are distinguished in Table 3, namely, those represented by reactions 13–14. The gammas produced by excitation of higher levels occur only for neutrons of energy above 10 MeV, and the cross sections for occurrence are only a few millibarns. Therefore, to simplify the calculation, these gamma rays have been replaced by two photons of energies 3.8 MeV and 4.8 MeV which provide approximately the correct yield of photon energy per disintegration. A similar grouping of cascade photons is represented by reaction 5. For purposes of calculation, a 2 value has been included in Table 3 for each of these averaged reactions 5, 15, 16. Of the remaining interactions, some involve absorption of the low energy neutron with production of a photon or a charged particle (e.g., 20 and 31 as those contributing most to dose), and others are nuclear interactions at rather high neutron energies (e.g., the (n,α) interactions such as 17 or 21–30). The energies of the particles following an interaction of these latter types were computed by the formula

$$E_2(\cos\theta) = \bar{E}_2 + E_1\left(2\frac{M_0M_1M_2M_3}{M^4}\left(1 + \frac{Q}{E_A}\right)\right)\cos\theta$$

where

M_0 = mass of target atom
M_1 = mass of incident particle
M_2 = mass of reaction product with energy E_2
M_3 = mass of the other reaction product
$M = M_0 + M_1 = M_2 + M_3$
Q = Q value for reaction

$$E_A = \frac{M_0}{M_0 + M_1} E_1$$

TABLE 3—*Reactions in tissue and associated energies*

Number	Interaction	Q Value	Particles and Photons	Photon Energy
		MeV		MeV
1	H(n,n)H			
2	C(n,n)C			
3	N(n,n)N			
4	O(n,n)O			
5	C(n,n')*C;	−1.75	$^*C \rightarrow {}^* C + \gamma_1$	1.75
6	C(n,n')*C;	−4.43	$^*C \rightarrow C + \gamma_2$	4.43
7	C(n,n')*C;	−6.8	$^*C \rightarrow C + \gamma_3$	6.8
8	N(n,n')*N;	−1.63	$^*N \rightarrow N + \gamma_1$	1.63
9	N(n,n')*N;	−2.31	$^*N \rightarrow N + \gamma_2$	2.31
10	N(n,n')*N;	−5.1	$^*N \rightarrow N + \gamma_3$	5.1
11	N(n,n')*N;	−10.0	$^*N \rightarrow N + \gamma_4$	10.0
12	N(n,n')*N;	−11.0	$^*N \rightarrow N + \gamma_5$	11.0
13	O(n,n')*O;	−6.1	$^*O \rightarrow O + \gamma_1$	6.1
14	O(n,n')*O;	−7.12	$^*O \rightarrow O + \gamma_2$	7.12
15	O(n,n')*O;	−3.8	$^*O \rightarrow {}^* O + \gamma_3$	3.8
16	O(n,n')*O;	−4.8	$^*O \rightarrow {}^* O + \gamma_4$	4.8
17	C(n,α_2)*Be;	−8.4	$^*Be \rightarrow Be + n \rightarrow 2\alpha + n$	
18	C(n,n')*C;	−10.0	$^*C \rightarrow Be + \alpha$	
19	N(n,2n)N	−10.6		
20	H(n,γ)H	2.23		2.23
21	C(n,α_0)Be	−5.7		
22	N(n,α_0)B	−0.75		
23	O(n,α_0)C	−2.2		
24	C(n,α_1)*Be;	−7.45	$^*Be \rightarrow Be + \gamma$	1.75
25	N(n,α_1)*B;	−2.25	$^*B \rightarrow B + \gamma_1$	2.1
26	N(n,α_2)*B;	−4.65	$^*B \rightarrow B + \gamma_2$	4.5
27	N(n,α_3)*B;	−5.15	$^*B \rightarrow B + \gamma_3$	5.0
28	O(n,α_1)*C;	−5.3	$^*C \rightarrow C + \gamma_1$	3.1
29	O(n,α_2)*C;	−6.0	$^*C \rightarrow C + \gamma_2$	3.8
30	O(n,α_3)*C;	−9.2	$^*C \rightarrow C + \gamma_3$	7.0
31	N(n,p)*C	0.628		
32	O(n,p)*N;	−9.6	$^*N \rightarrow N + \gamma$	6.1
33	N(n,t)C	−4.0		

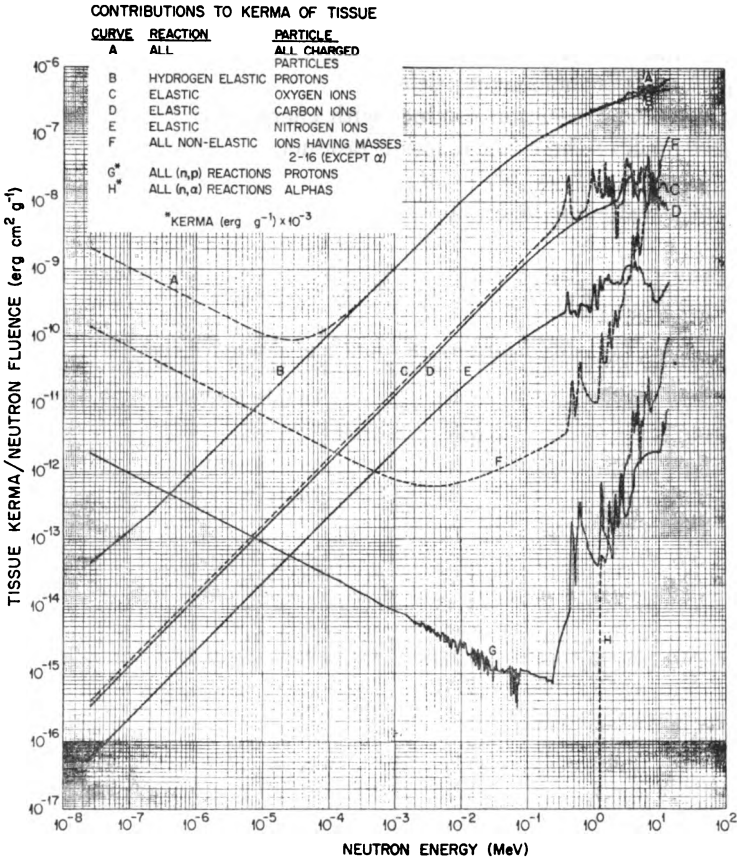

Fig. 3. Contributions to tissue kerma.

$$\bar{E}_2 = \frac{2M_1M_2E_1}{(M_2 + M_3)^2} + \frac{M_3Q + E_1(M_3 - M_1)}{(M_2 + M_3)} = \text{average energy of reaction product with mass } M_2$$

E_1 = energy of incident particle

θ = angle between the exit particle and the entrance trajectory in the center of mass system.

This procedure conserves both energy and momentum. When neutrons were produced, they were followed independently thereafter in the calculation.

The cross section data for these interactions were taken largely from BNL-325 (BNL, 1958 and BNL, 1964) and the ORNL 05R Cross Section Library,[8] but had to be interpolated to fill in gaps in the data. The

[8] See Irving et al., 1965.

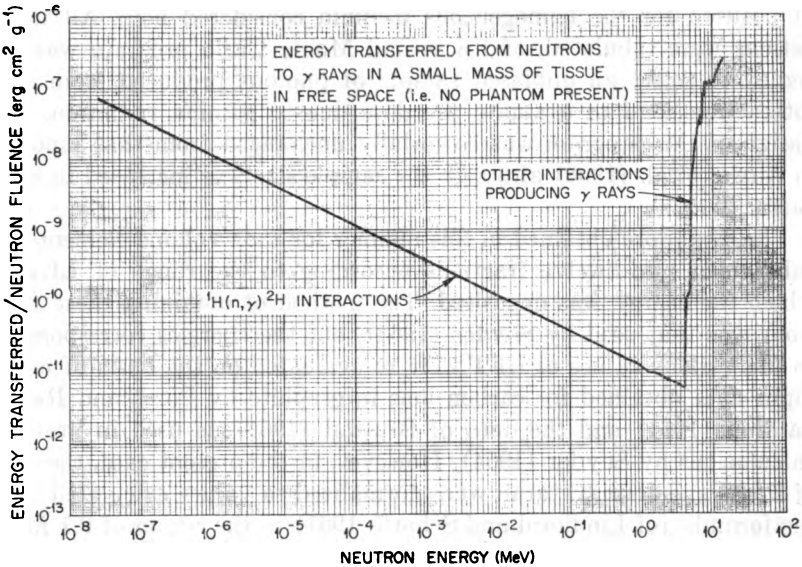

Fig. 4. Energy transferred from neutrons to γ rays in a small mass of tissue in free space (i.e., no phantom present).

macroscopic cross sections used differ somewhat from those employed by Auxier et al. (1968) because of the utilization of more recent results and other adjustments, but these changes are unlikely to significantly alter the results.

Using these cross sections, the kerma for tissue for unit fluence of neutrons has been computed and is shown graphically in Figure 3. In cases where it was not convenient to compute the average energy of particles directly, a sample of interactions was obtained by use of the computer code, and the energy released was estimated from this sample. The energy released to photons has been estimated similarly and is shown in Figure 4.

The neutron "case histories" were computed in a straightforward manner except that no neutron was allowed to be absorbed; instead its weight (originally one) was multiplied after each interaction by a factor equal to its probability of survival of the interaction. The energy carried by each recoil nucleus (or by protons, alpha particles, and ^3H nuclei produced by the interaction) was considered to be absorbed in the volume element where the interaction occurred. This seems justified because the maximum range of any of these particles in tissue when produced by neutrons of energy 14 MeV does not exceed 0.25 cm and is much below this for lower neutron energies and also because the interaction density discussed above will be a slowly varying function

of position for the homogeneous medium considered here. All interactions were tabulated, and a second Monte Carlo program was designed to select an unbiased sample of photons produced from such interactions and to evaluate the dose due to gamma radiation. The design of this program was essentially unchanged from that reported in Auxier et al. (1968) although the sample size was increased in some cases.

The total dose obtained in this manner for each volume element was subdivided into twelve fractions according to the range of LET at which the energy was deposited. The method of obtaining these fractions was that used by Snyder (1964), but the energies corresponding to the various values of LET were re-evaluated on the basis of newer input data obtained for carbon ions in graphite by Porat and Ramavataran (1961) and the data of Brustad (1961) as used in previous calculations by Snyder (1964). However, in the present case, the data of Porat and Ramavataran were normalized to values calculated from the formula (of Lindhard and Scharff, 1961) in the region of 0.7 MeV, and the low-energy tail was based on Northcliffe's extrapolation (Northcliffe, 1963). The energies at which the various particles have LET values corresponding to the boundaries are shown in Table 4 as well as the values of QF considered to be appropriate for these values of LET. These values are either those recommended by NCRP Report No. 17 (NCRP, 1954) or are derived from them by interpolation. The total energy deposited in a volume element with LET in the various ranges was tabulated and weighted with the QF factors shown in Table 4. These weighting factors were taken as the average of the QF values for the values of LET corresponding to the ends of the range.

Only results for broad beams of monoenergetic neutrons are reported here. In each case, the beam is parallel to the base of the cylinder, and at least 20,000 neutrons were used for the sample. The average dose and dose equivalent are shown in Figures 5 through 34. The dose due to photons and the dose due to charged particles are shown separately, and similarly for dose equivalent. Only half of the lateral distribution is shown because, by symmetry, the other half would be the same.

It did not seem feasible to indicate on the dose curves the standard deviation (SD), σ, for each point. An interval representing $\pm$ SD has been shown whenever the SD exceeded 10 percent. It will be noted that at the greater depths and for the lower energies, the statistical accuracy is less, as might be expected because relatively few neutrons penetrate to these regions of the phantom. It should be noted that the SD is sometimes 50 percent or more of the mean dose or dose equivalent in these regions. In such cases it is unlikely that the distribution is approxi-

TABLE 4—*Energies*[a] E_1 *and* E_2 *(MeV) for specified values of LET and values of QF for intervals of LET*

LET	QF[b]	Hydrogen		Tritium		Helium		Beryllium		Boron		Carbon		Nitrogen		Oxygen	
		E_1	E_2	E_1	E_2	E_1	E_2	E_1	E_2	E_1	E_2	E_1	E_2	E_1	E_2	E_1	E_2
keV/μm																	
	1.00																
3.5		.0001	13.50	.0002	41.00	.00004	500.0	.000055	100.	.000044	100.	.000042	100.	.000038	100.	.000036	100.
	1.50																
7.0		.0004	6.00	.0008	18.50	.00018	125.0	.000215	100.	.00018	100.	.000165	100.	.000160	100.	.000150	100.
	2.82																
15.0		.0017	2.30	.0036	6.90	.00086	50.0	.000960	100.	.00086	100.	.000740	100.	.00074	100.	.000700	100.
	4.47																
25.0		.0046	1.15	.0100	3.50	.0025	27.0	.00260	100.	.00240	100.	.002100	100.	.00210	100.	.00200	100.
	6.18																
35.0		.0090	0.72	.0200	2.10	.0052	17.5	.00520	100.	.00470	100.	.00420	100.	.00415	100.	.00400	100.
	8.28																
50.0		.0180	0.42	.0400	1.25	.0110	11.0	.0105	100.	.00970	100.	.00880	100.	.00820	100.	.00700	100.
	10.30																
62.5		.0275	0.28	.0620	0.90	.0170	8.4	.0165	100.	.0150	100.	.01350	100.	.01300	100.	.0125	100.
	11.80																
75.0		.0390	0.20	.0900	0.63	.0250	6.5	.0240	100.	.0215	100.	.01950	100.	.01850	100.	.0175	100.
	13.60																
87.5		.0520	0.15	.1200	0.44	.0360	5.2	.0320	100.	.0295	100.	.0265	100.	.02500	100.	.0240	100.
	14.90																
100.0		.0850	0.085	.2500	0.25	.0480	4.2	.0440	100.	.0400	100.	.0350	100.	.03300	100.	.0310	100.
	17.50																
200.0		—	—	—	—	.2500	1.0	.1850	100.	.1550	100.	.1350	100.	.1300	100.	.2000	100.
	20.0																
950.0		—	—	—	—	—	—	—	—	—	—	—	—	—	—	—	—

[a] The charged particles have the same LET at two energies that are below and above the maximum of the Bragg curve.

[b] Values of QF are staggered to indicate values chosen for the respective intervals LET (e.g., QF = 1 up to 3.5 keV/μm, 1.5 between 3.5 keV/μm to 7.0 keV/μm etc.).

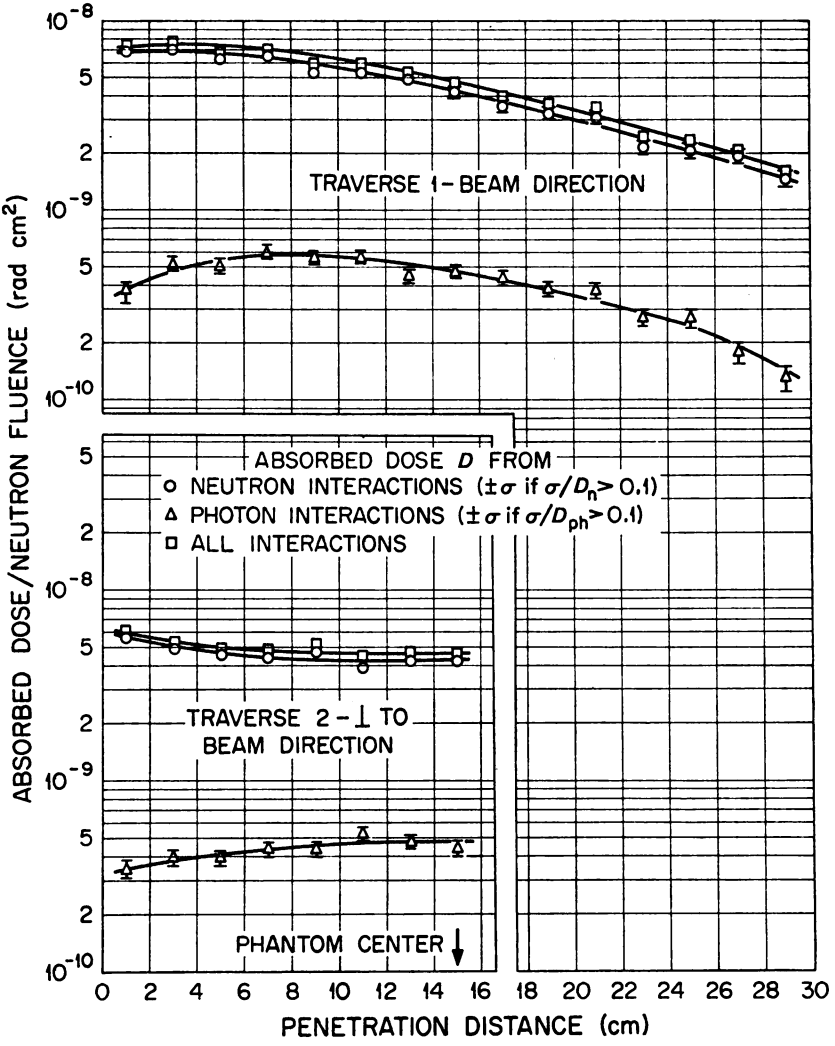

Fig. 5. Absorbed dose in central region of the cylindrical phantom. (Neutron energy 14 MeV.)

mately normal, and the use of the standard deviation to define a confidence interval for the mean as for normal distributions is unjustified. Also, it should be recognized that the standard deviations reflect only the statistical variability inherent in the Monte Carlo method and are not influenced by systematic errors such as inaccuracies in the cross

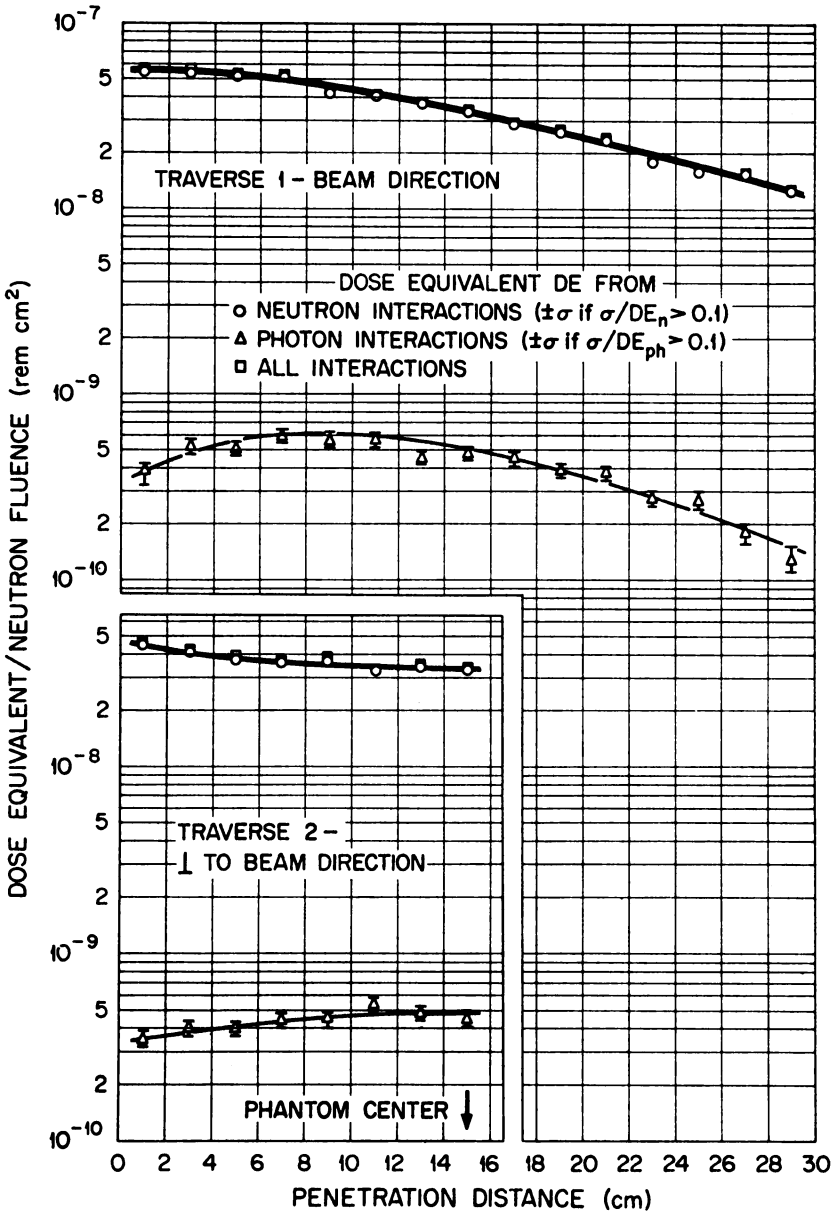

Fig. 6. Dose equivalent in central region of the cylindrical phantom. (Neutron energy 14 MeV.)

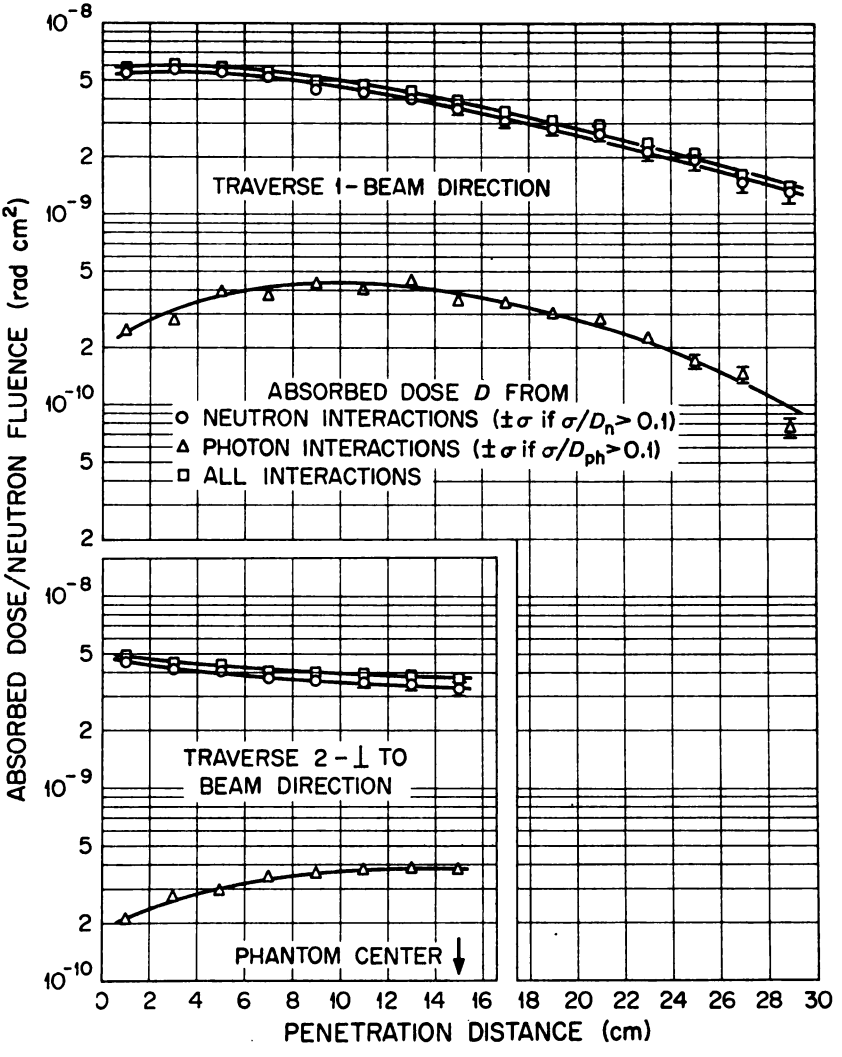

Fig. 7. Absorbed dose in central region of the cylindrical phantom. (Neutron energy 10 MeV.)

sections or simplifications in the programming of the physical interactions, etc. Moreover, the coefficients of variation for the dose due to photons represent only the statistical variability of the Monte Carlo calculation of the photon source programmed. However, this photon source is a result of the neutron calculation and hence involves some

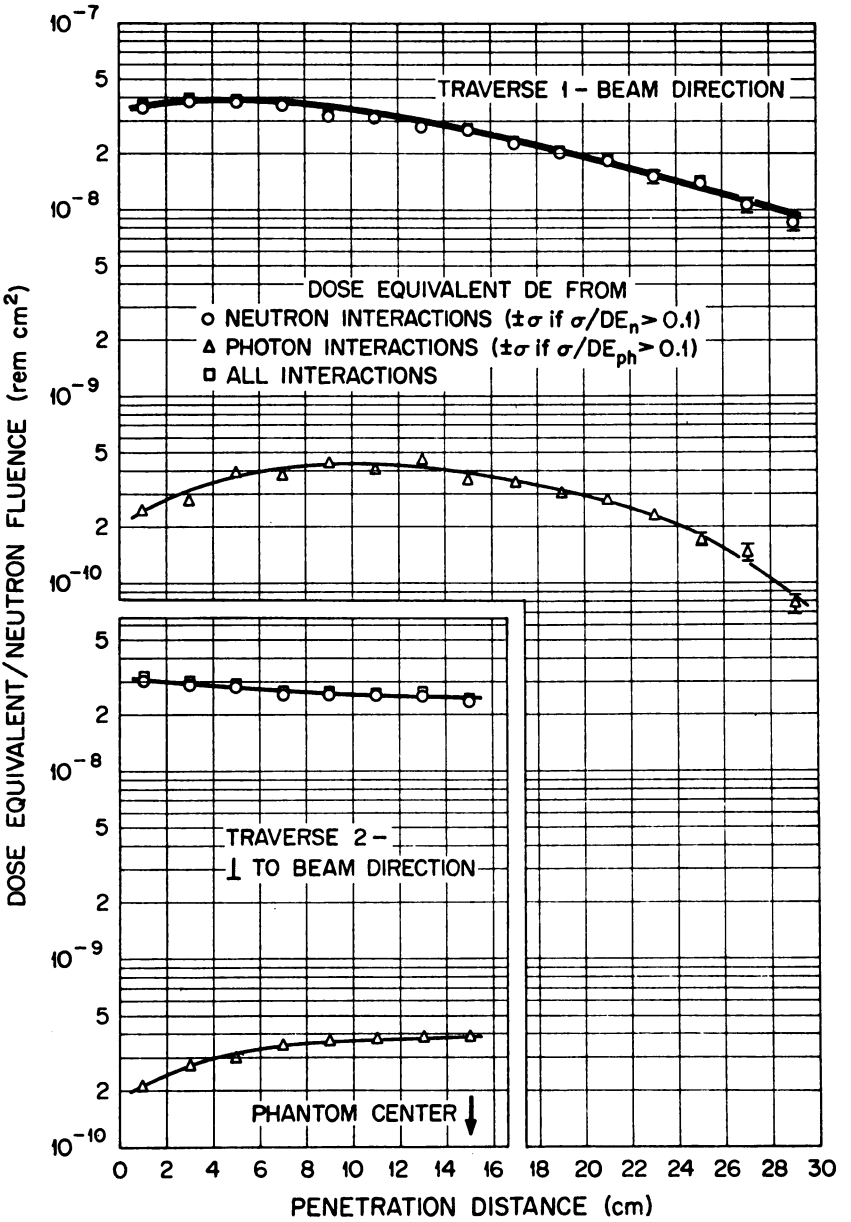

Fig. 8. Dose equivalent in central region of the cylindrical phantom. (Neutron energy 10 MeV.)

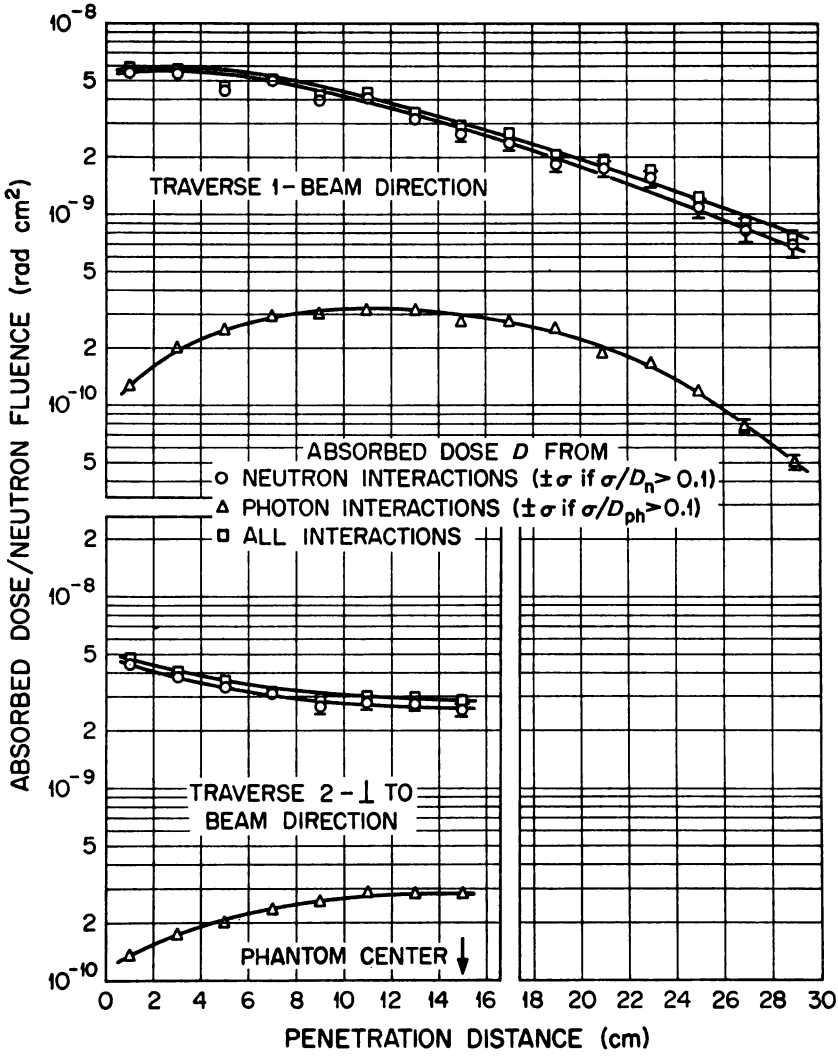

Fig. 9. Absorbed dose in central region of the cylindrical phantom. (Neutron energy 7 MeV.)

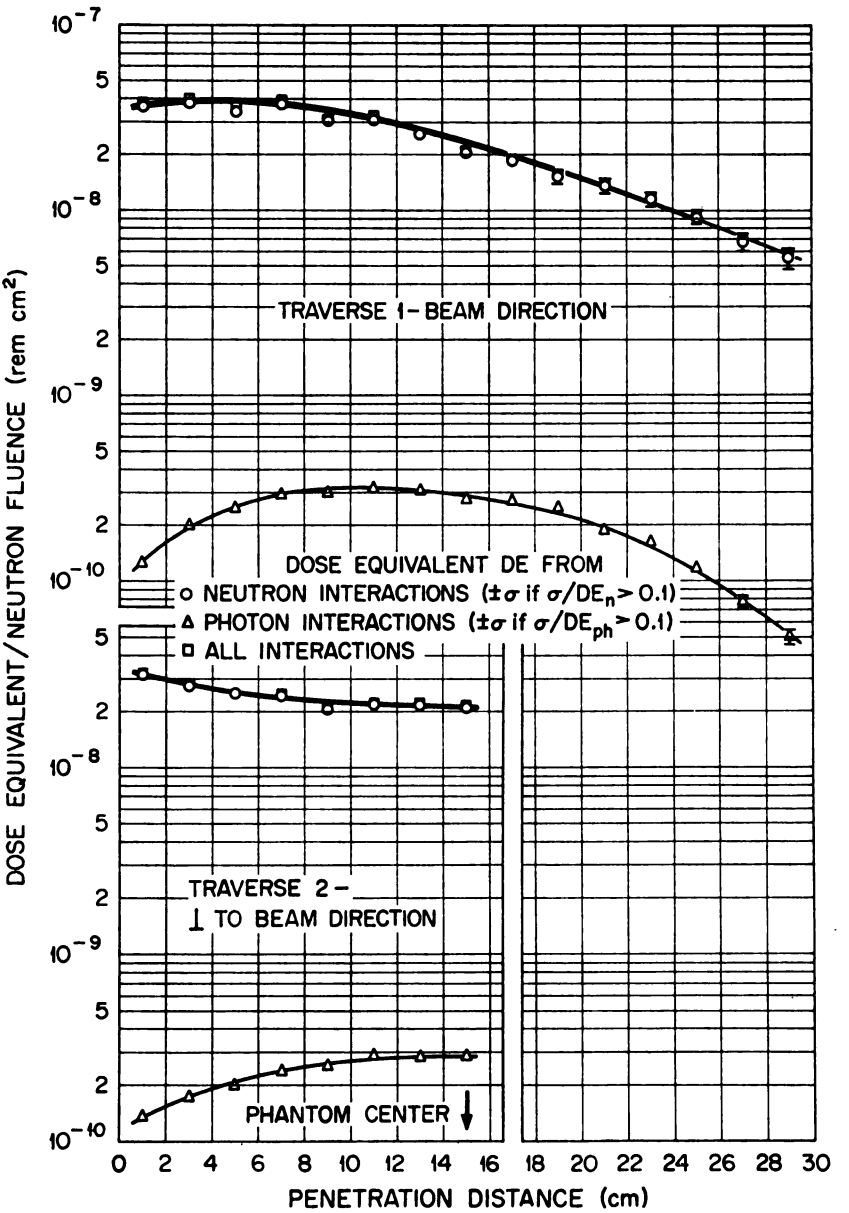

Fig. 10. Dose equivalent in central region of the cylindrical phantom. (Neutron energy 7 MeV.)

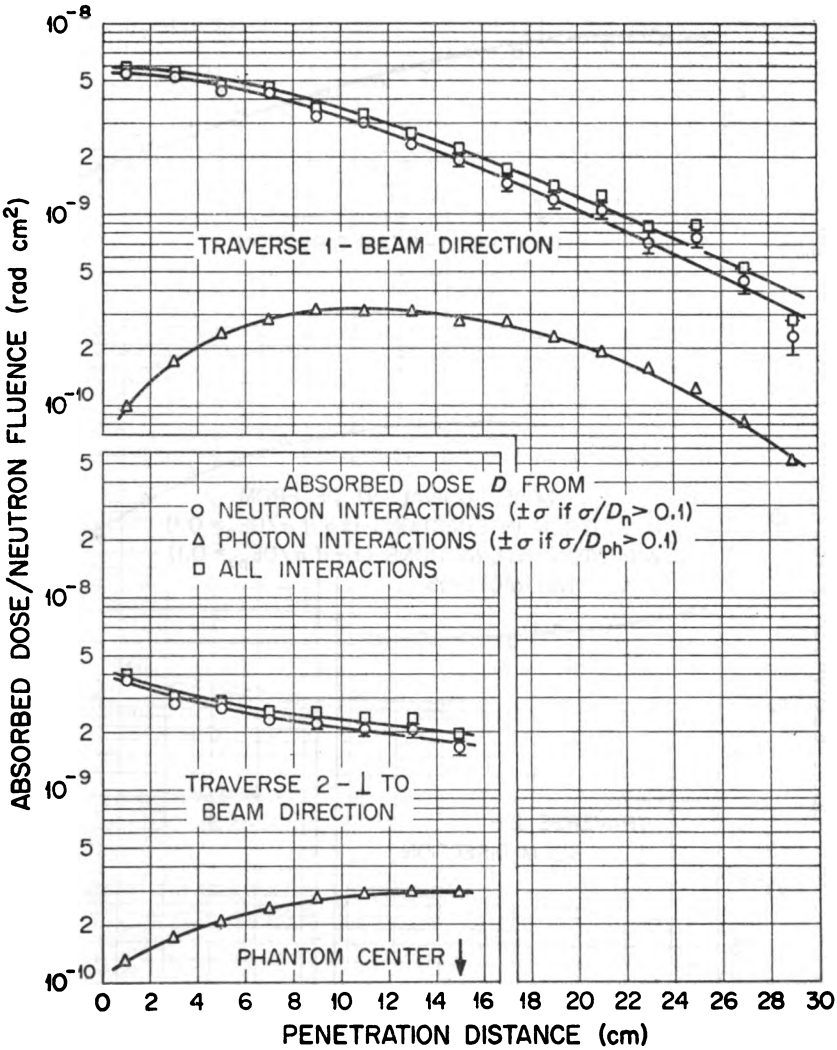

Fig. 11. Absorbed dose in central region of the cylindrical phantom. (Neutron energy 5 MeV.)

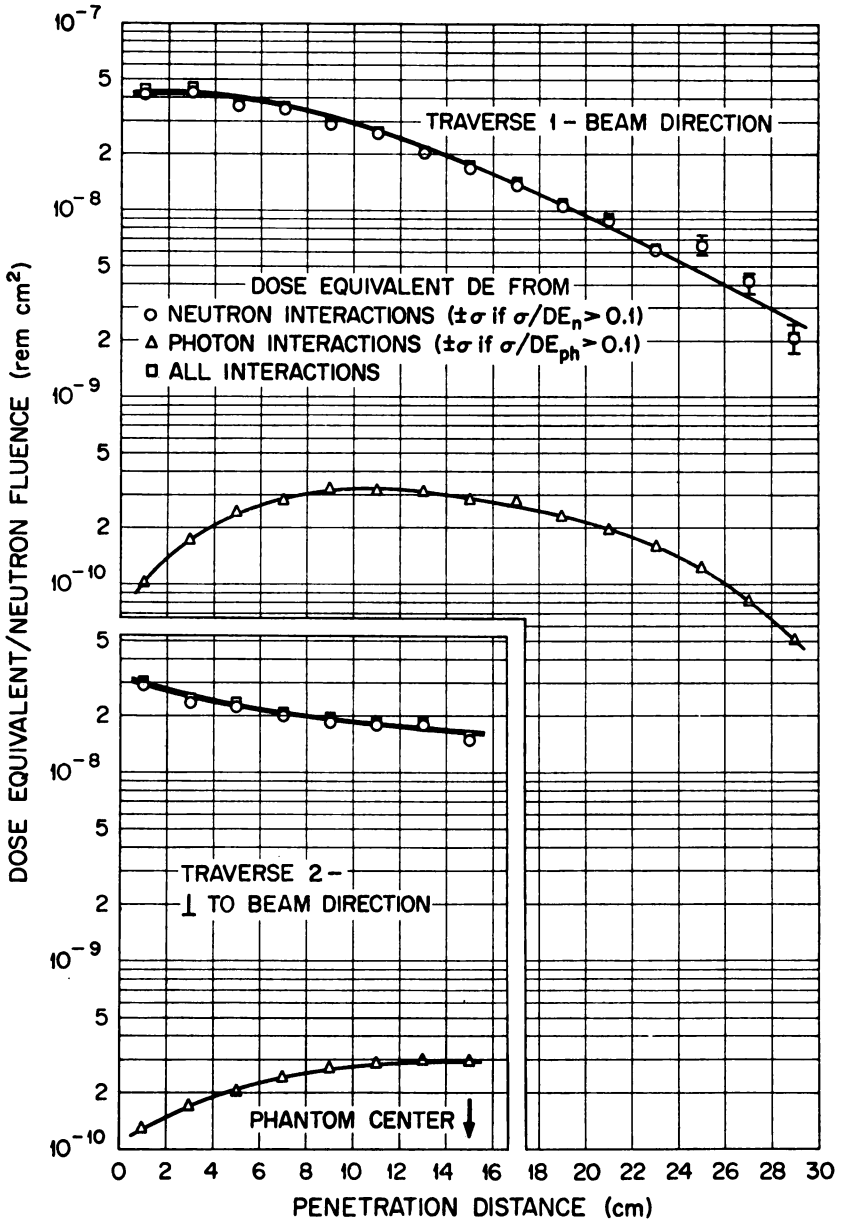

Fig. 12. Dose equivalent in central region of the cylindrical phantom. (Neutron energy 5 MeV.)

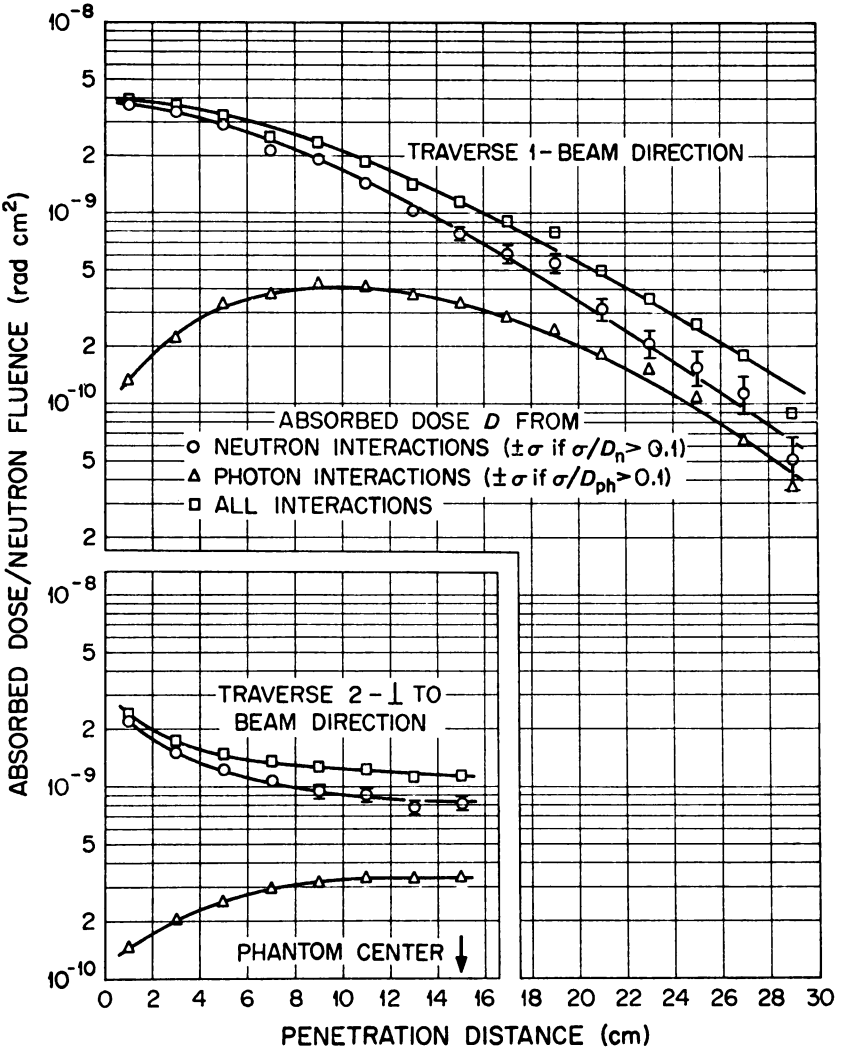

Fig. 13. Absorbed dose in central region of the cylindrical phantom. (Neutron energy 2.5 MeV.)

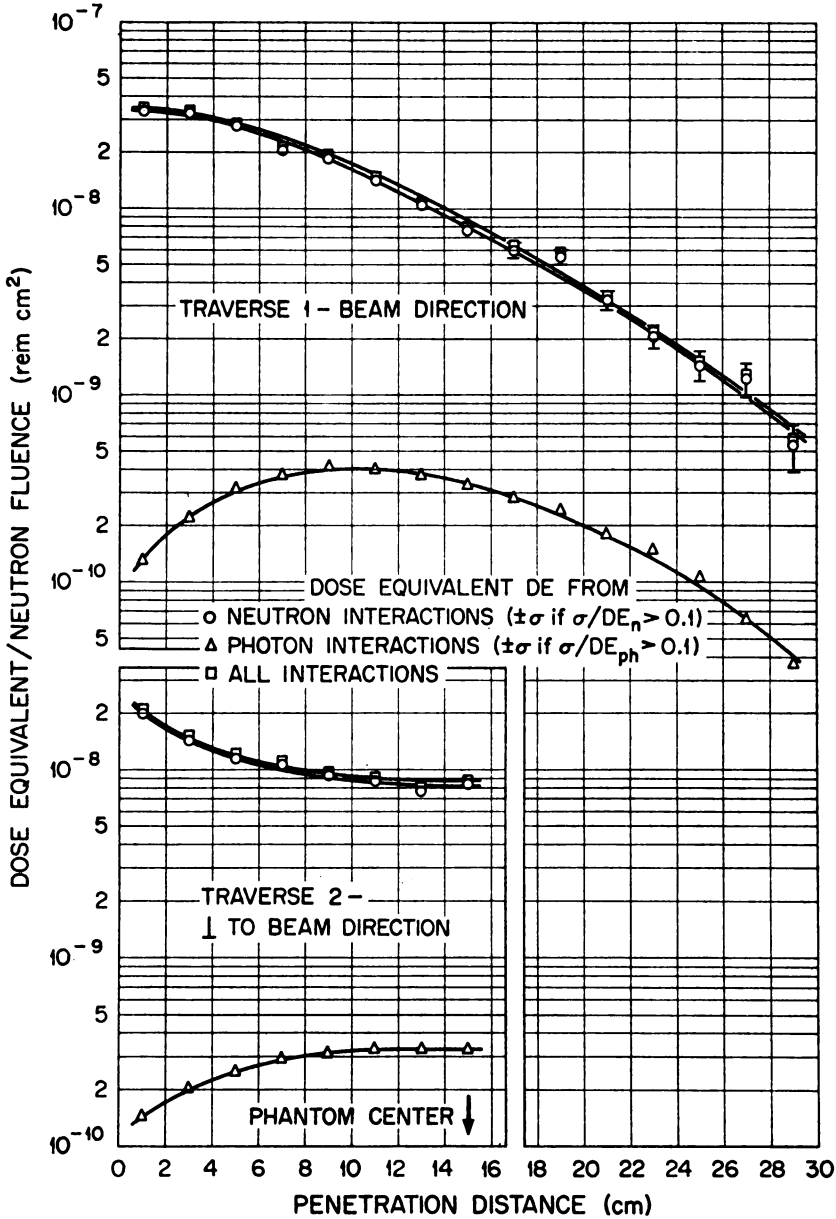

Fig. 14. Dose equivalent in central region of the cylindrical phantom. (Neutron energy 2.5 MeV.)

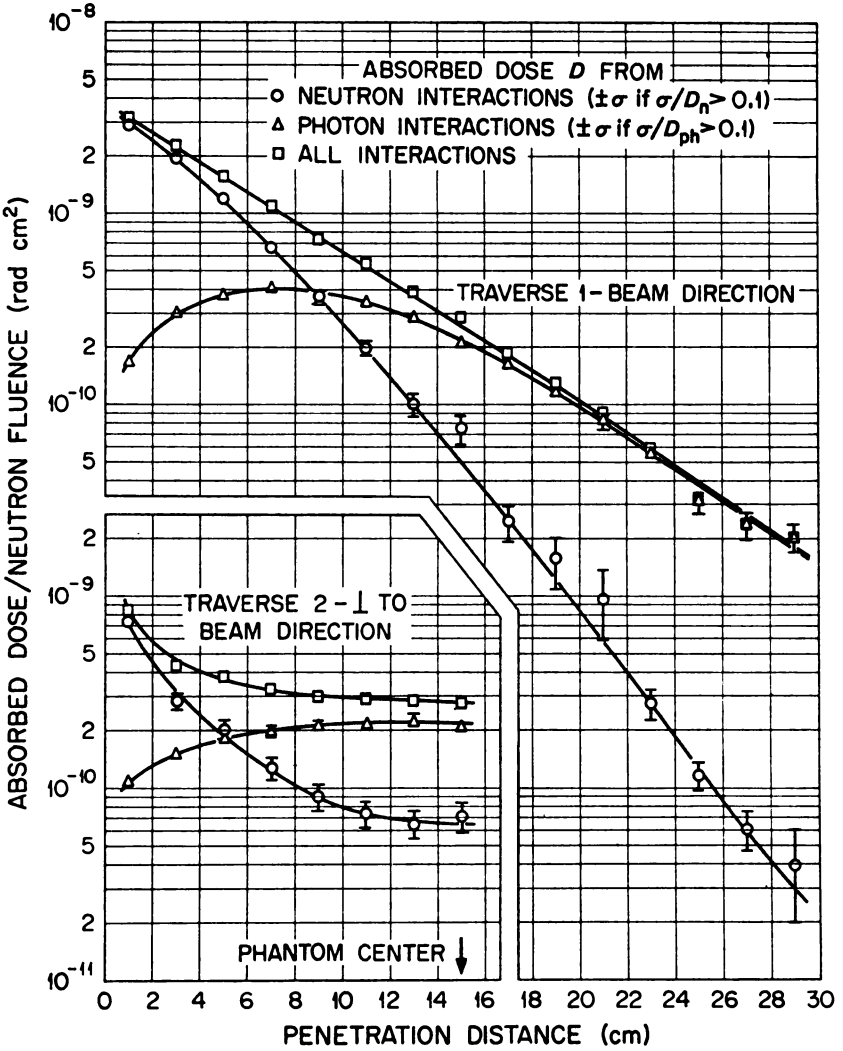

Fig. 15. Absorbed dose in central region of the cylindrical phantom. (Neutron energy 1 MeV.)

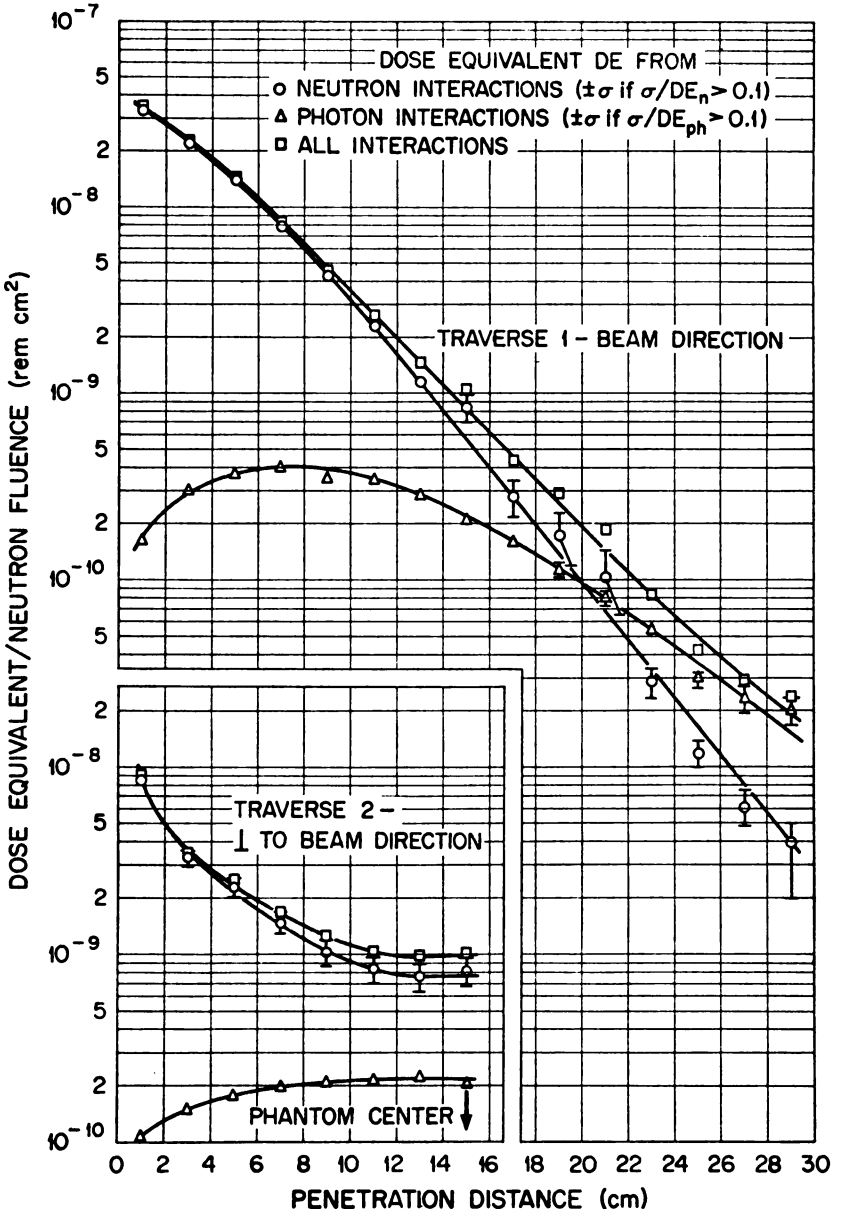

Fig. 16. Dose equivalent in central region of the cylindrical phantom. (Neutron energy 1 MeV.)

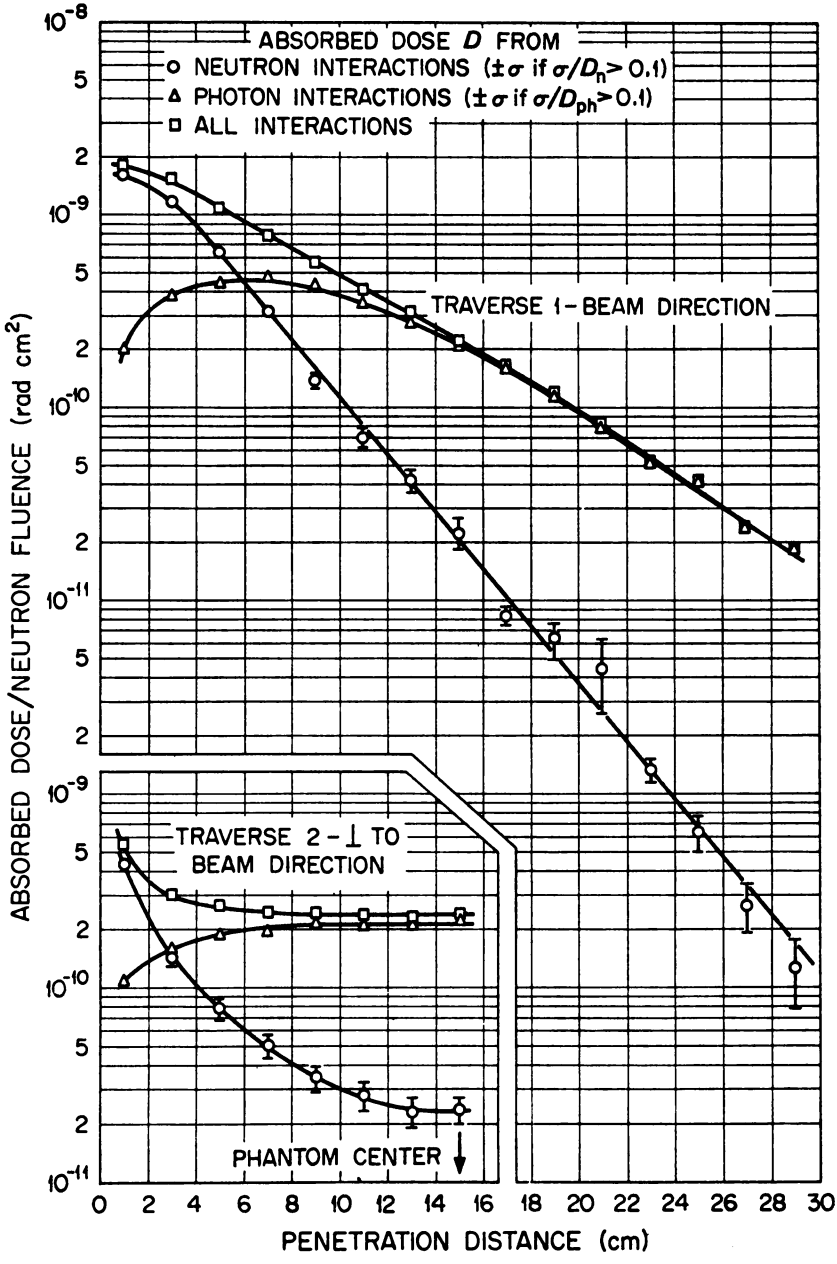

Fig. 17. Absorbed dose in central region of the cylindrical phantom. (Neutron energy 0.5 MeV.)

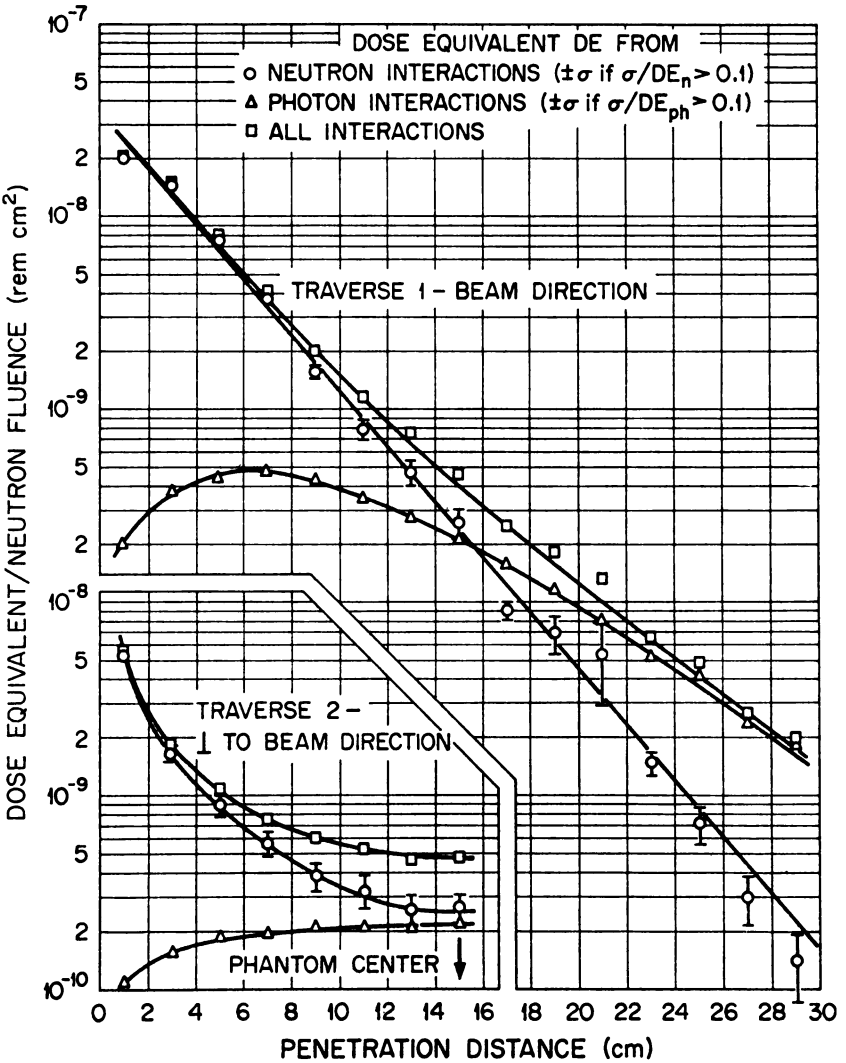

Fig. 18. Dose equivalent in central region of the cylindrical phantom. (Neutron energy 0.5 MeV.)

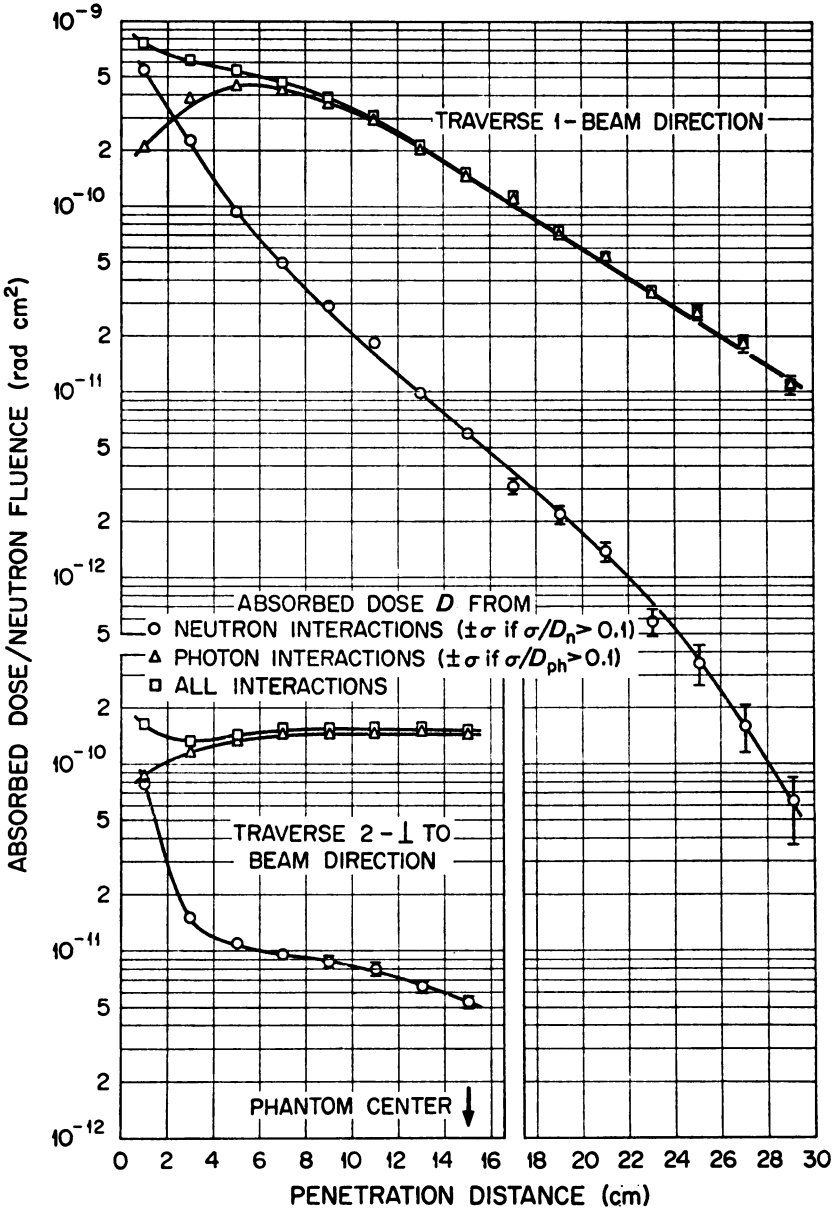

Fig. 19. Absorbed dose in central region of the cylindrical phantom. (Neutron energy 0.1 MeV.)

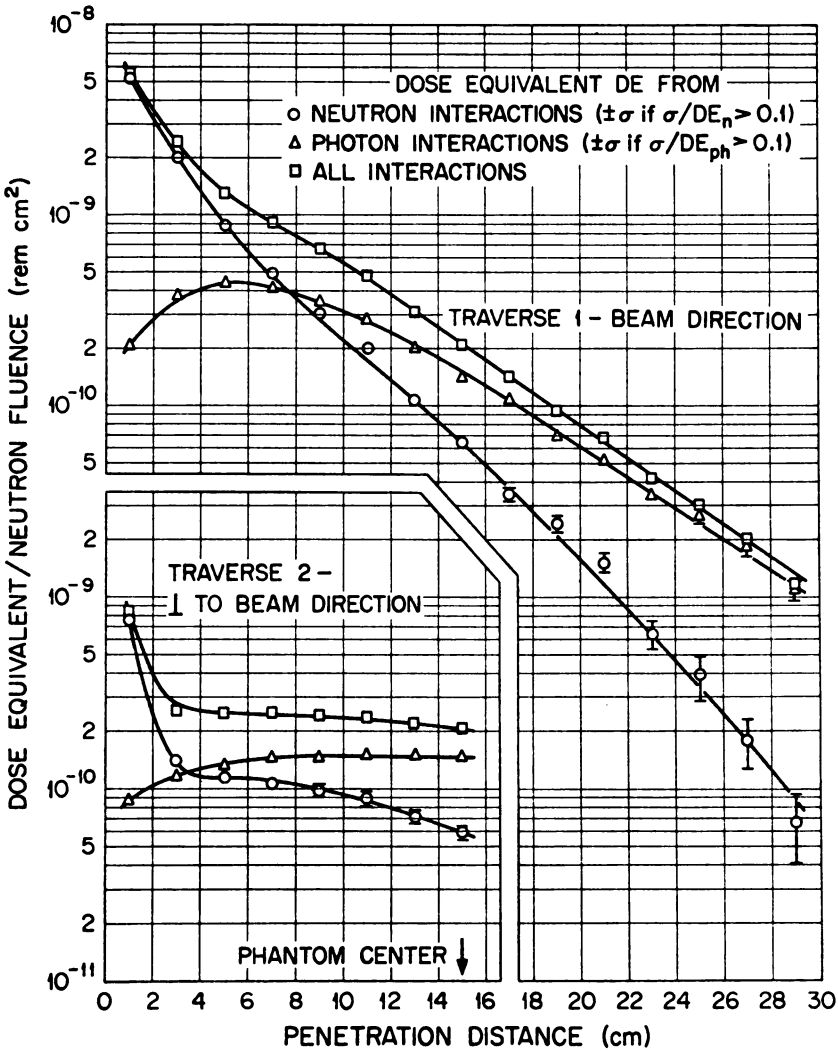

Fig. 20. Dose equivalent in central region of the cylindrical phantom. (Neutron energy 0.1 MeV.)

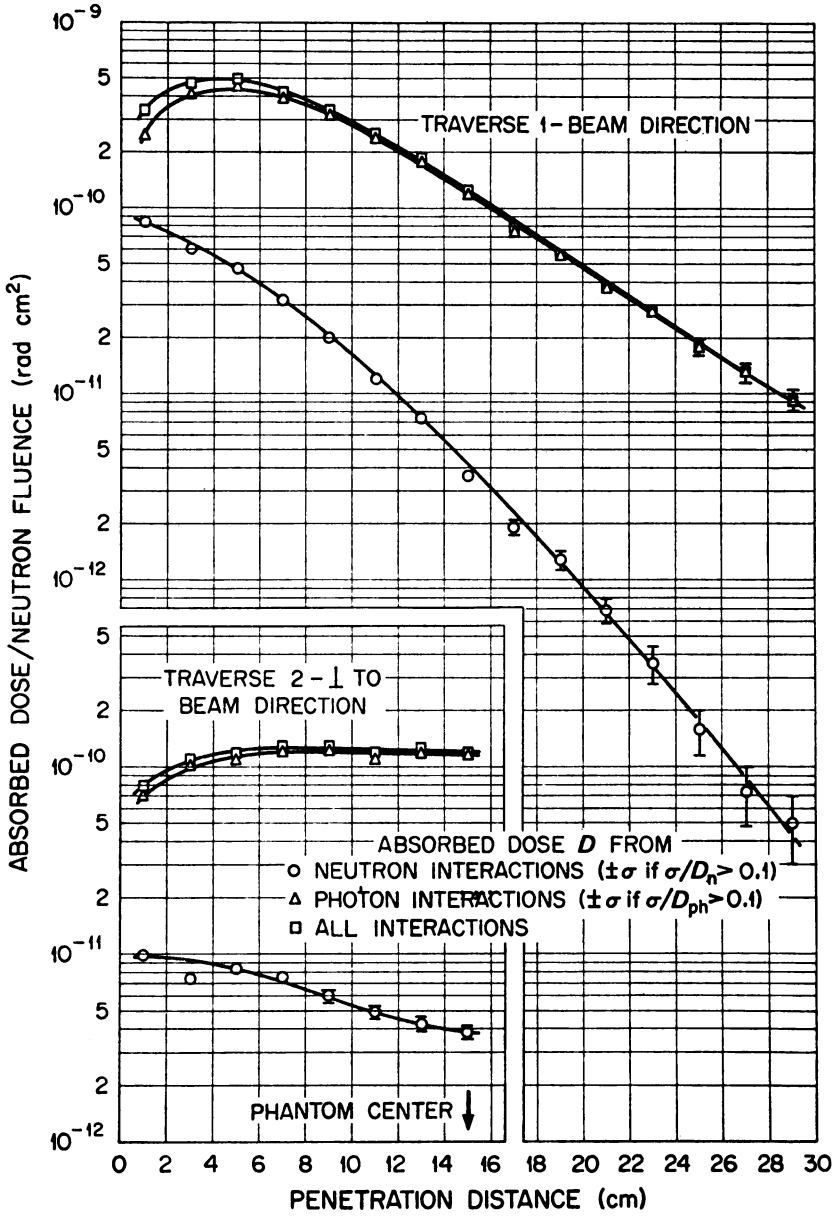

Fig. 21. Absorbed dose in central region of the cylindrical phantom. (Neutron energy 10 keV.)

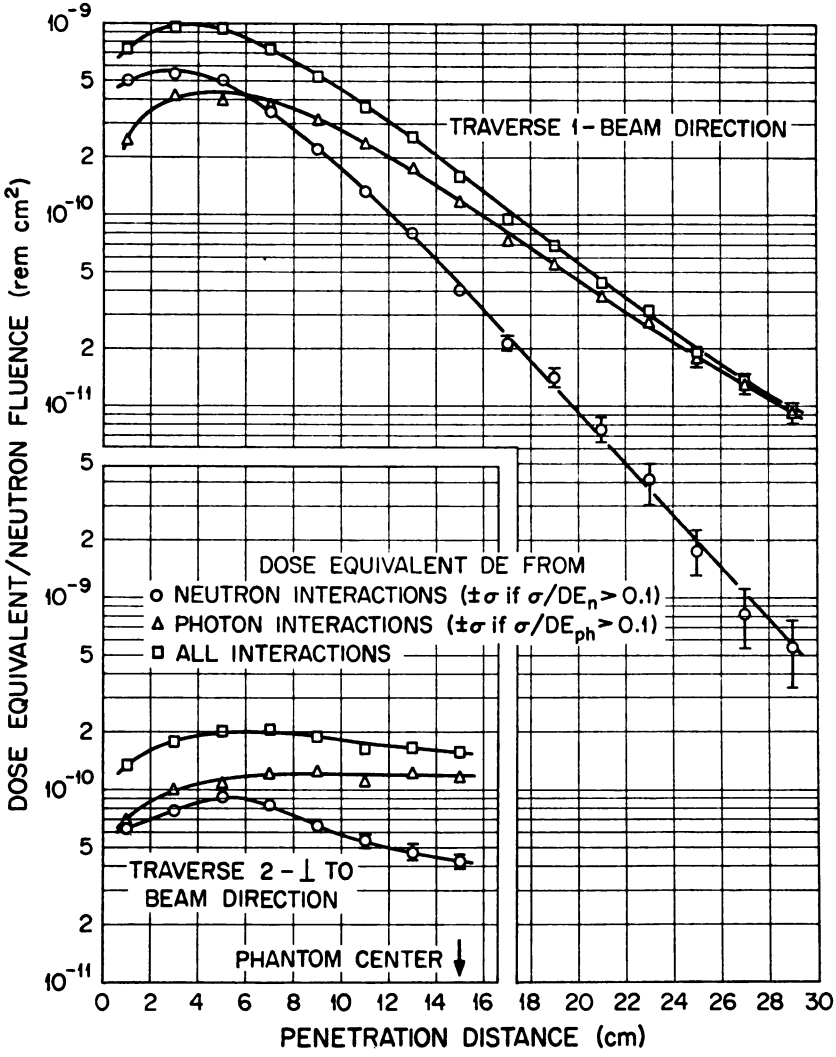

Fig. 22. Dose equivalent in central region of the cylindrical phantom. (Neutron energy 10 keV.)

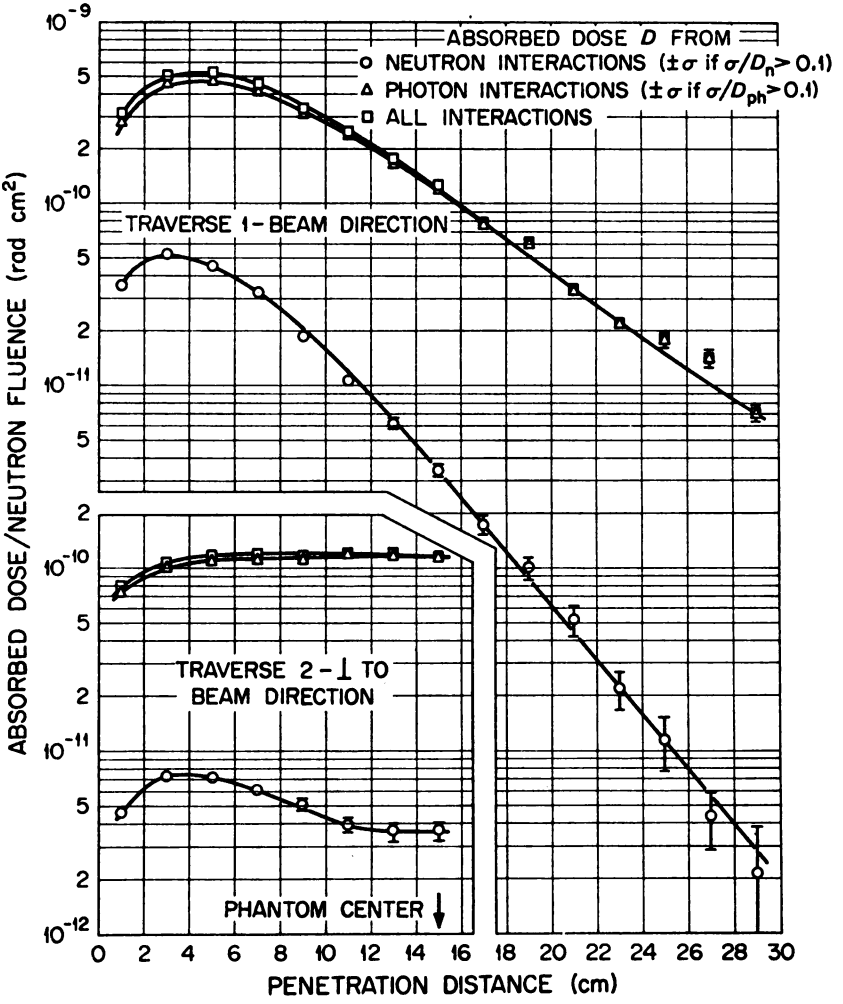

Fig. 23. Absorbed dose in central region of the cylindrical phantom. (Neutron energy 1 keV.)

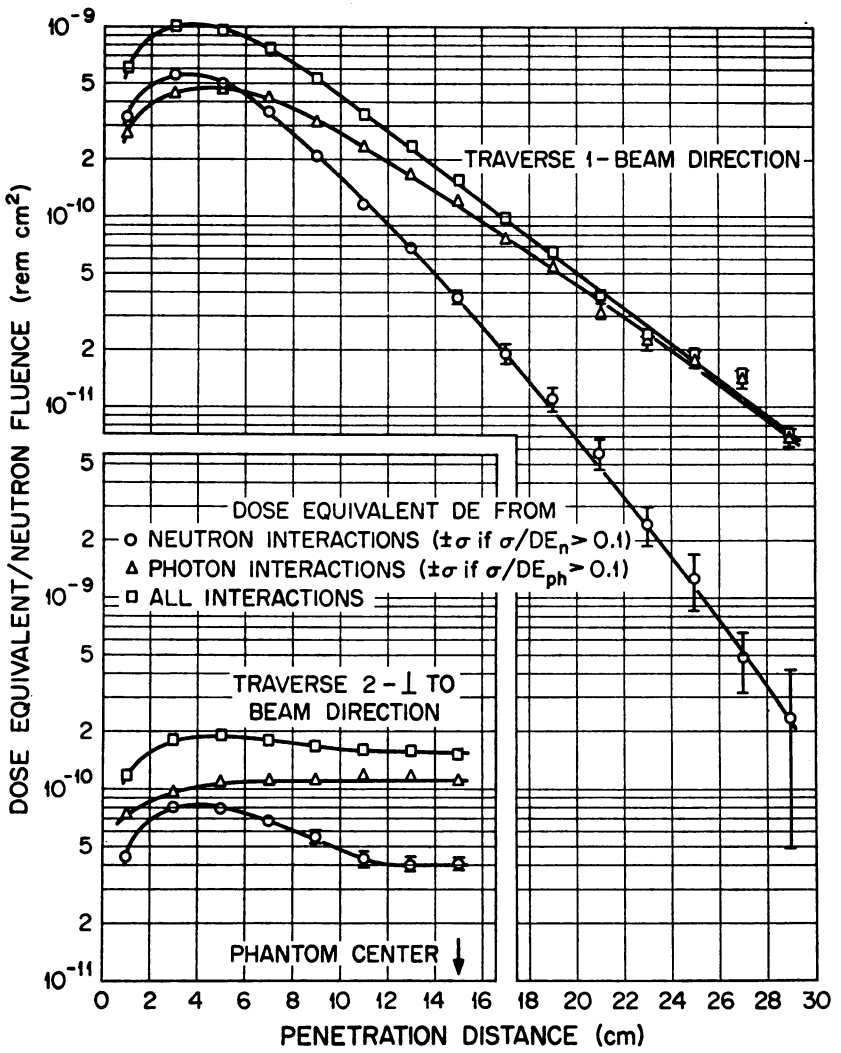

Fig. 24. Dose equivalent in central region of the cylindrical phantom. (Neutron energy 1 keV.)

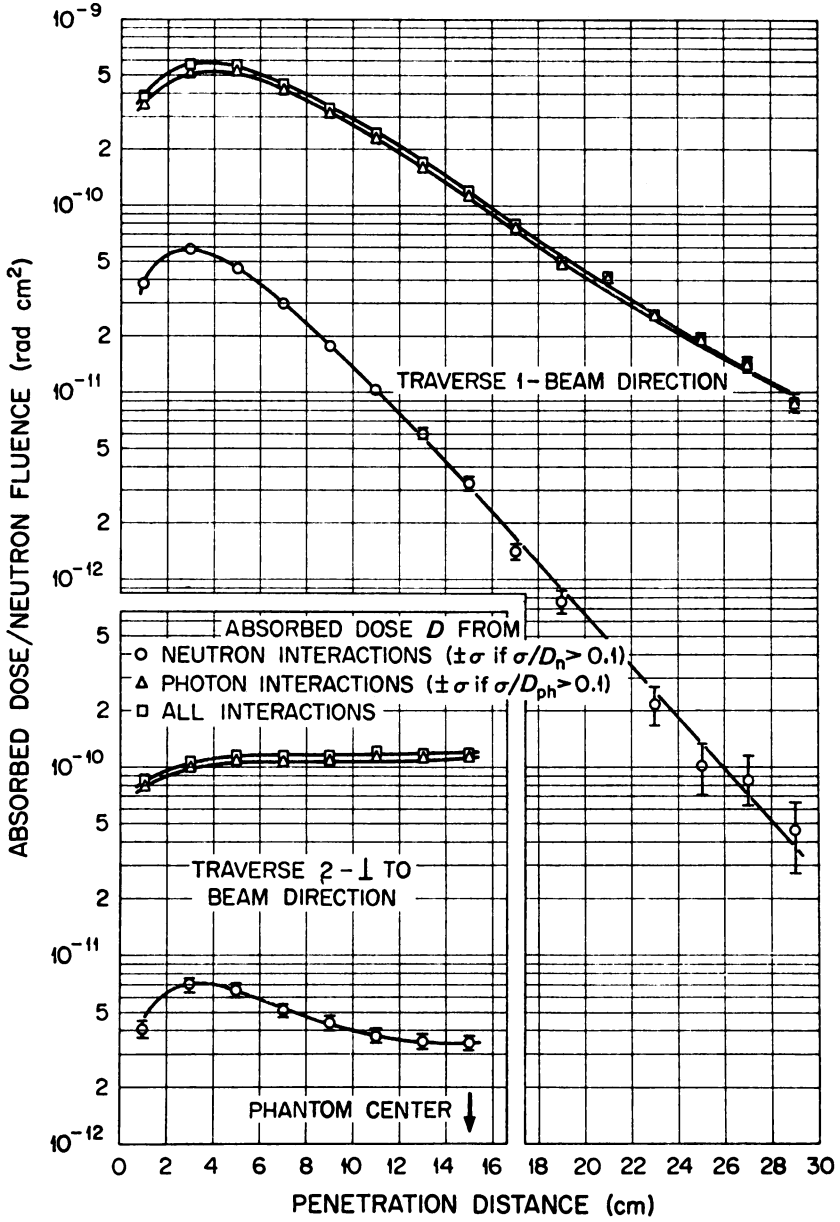

Fig. 25. Absorbed dose in central region of the cylindrical phantom. (Neutron energy 100 eV.)

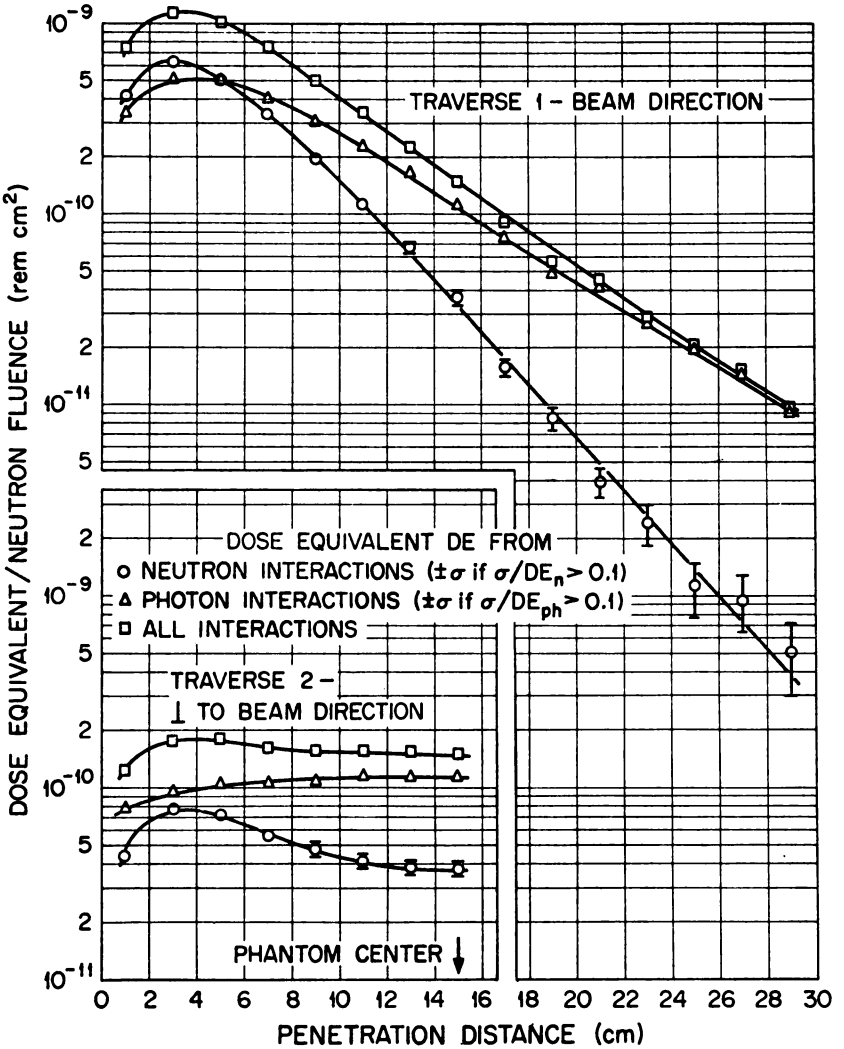

Fig. 26. Dose equivalent in central region of the cylindrical phantom. (Neutron energy 100 eV.)

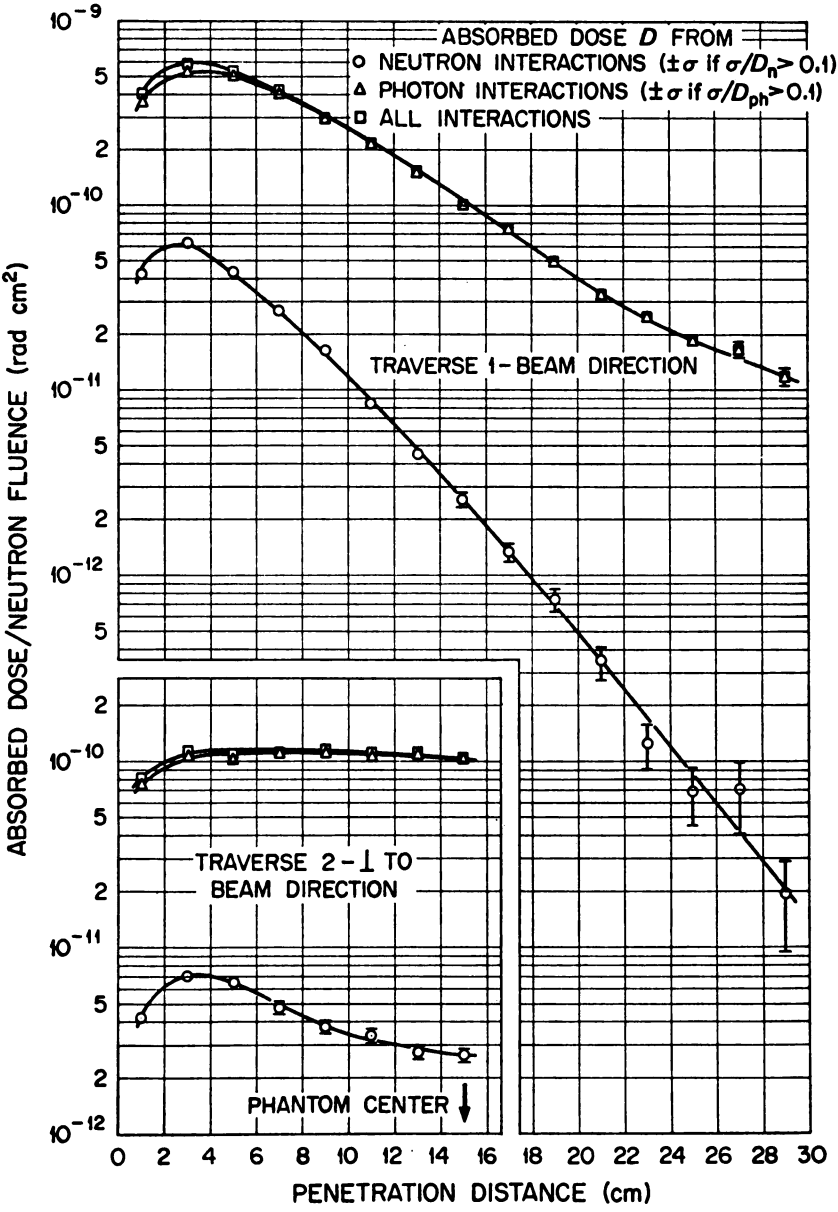

Fig. 27. Absorbed dose in central region of the cylindrical phantom. (Neutron energy 10 eV.)

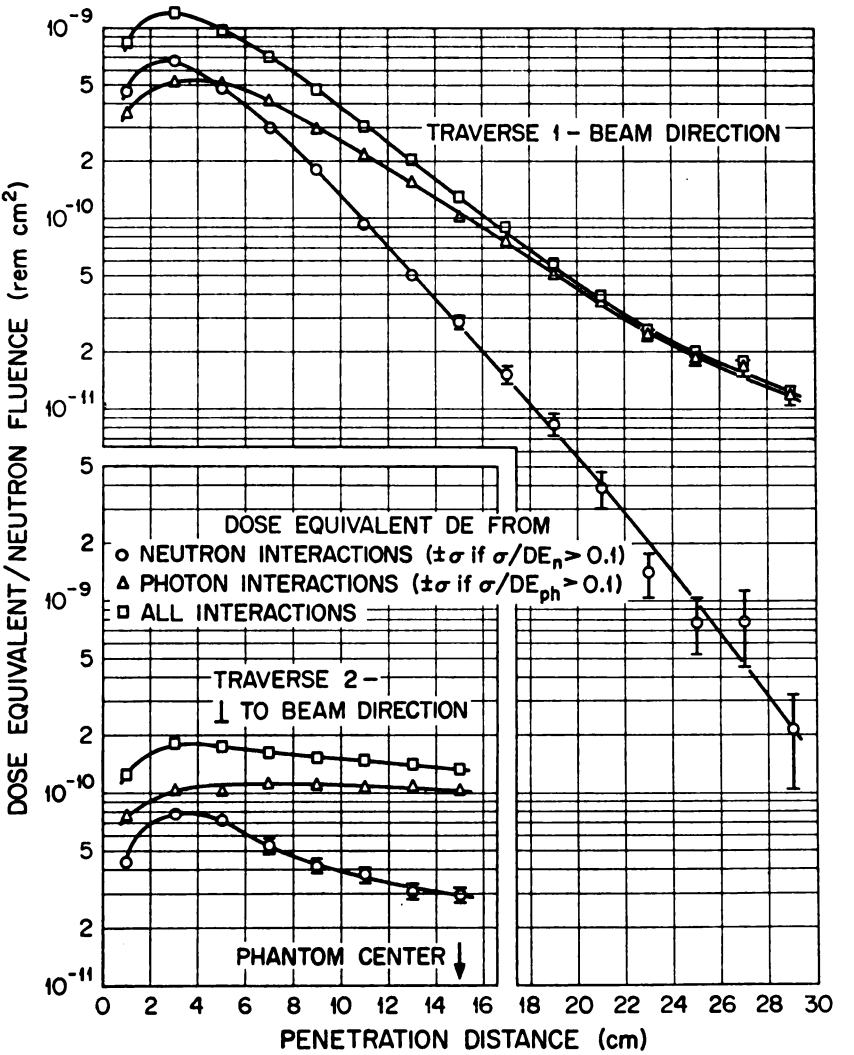

Fig. 28. Dose equivalent in central region of the cylindrical phantom. (Neutron energy 10 eV.)

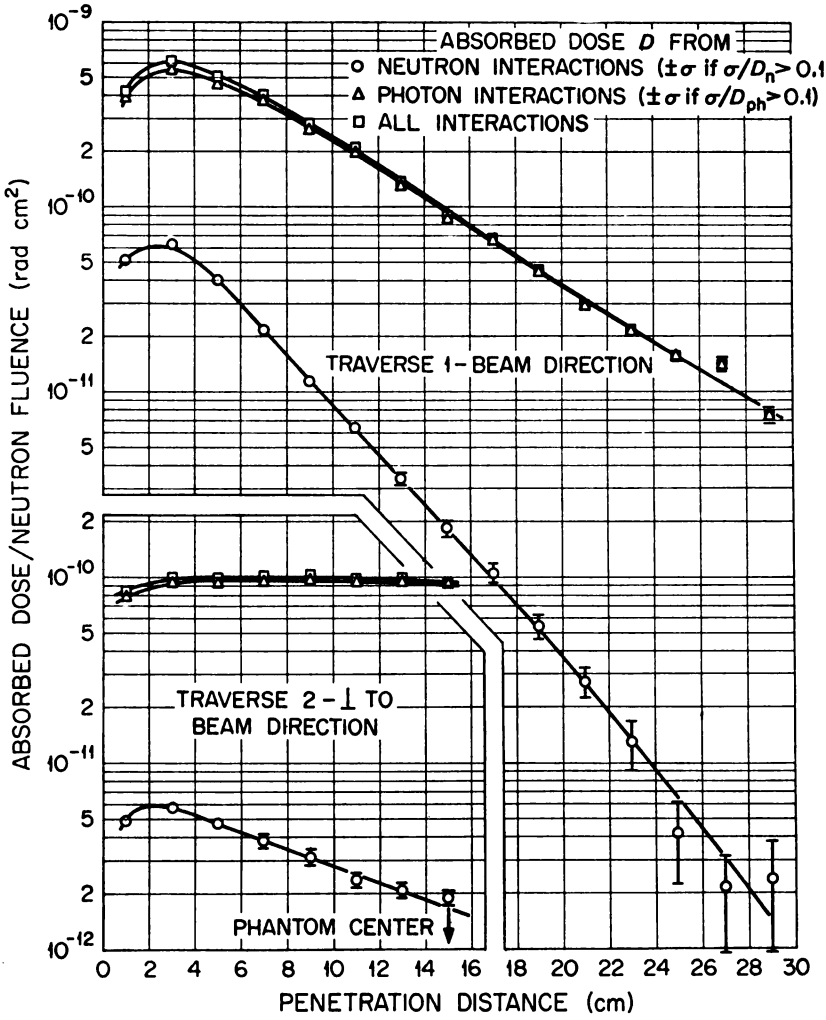

Fig. 29. Absorbed dose in central region of the cylindrical phantom. (Neutron energy 1 eV.)

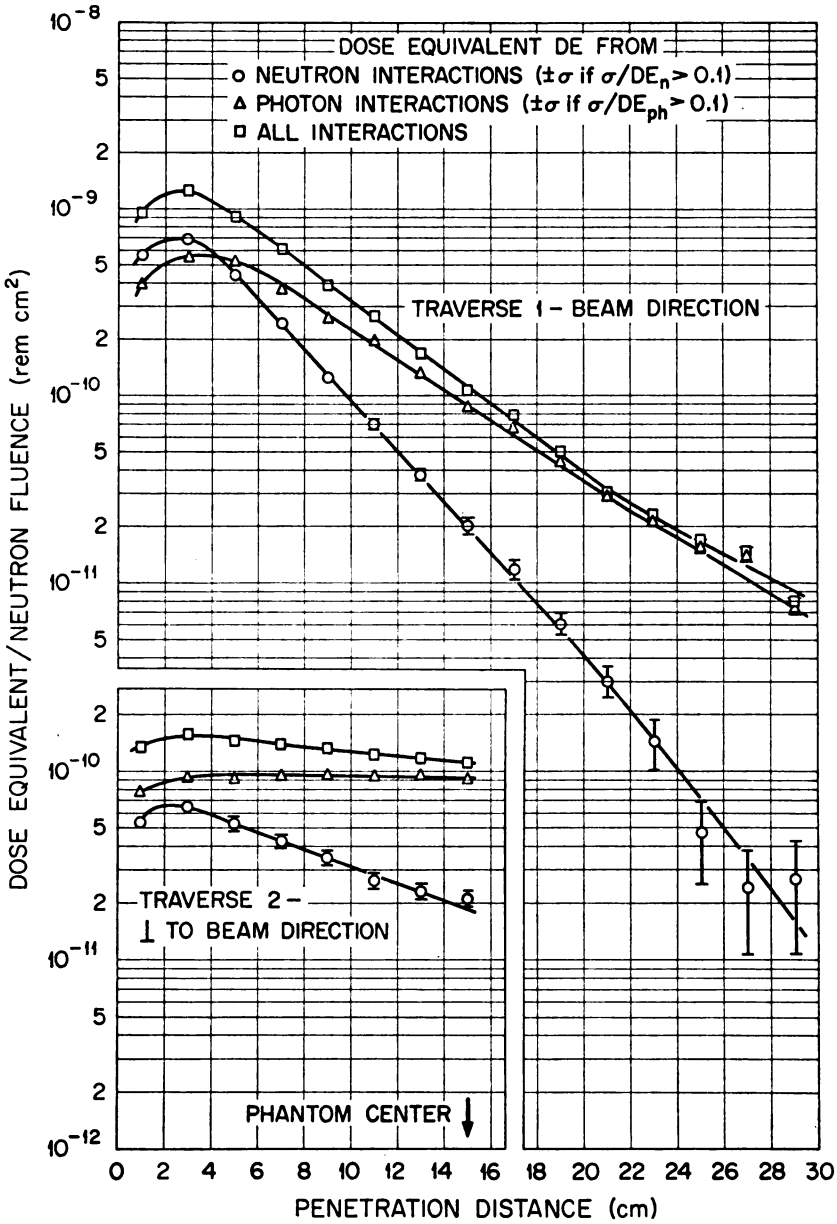

Fig. 30. Dose equivalent in central region of the cylindrical phantom. (Neutron energy 1 eV.)

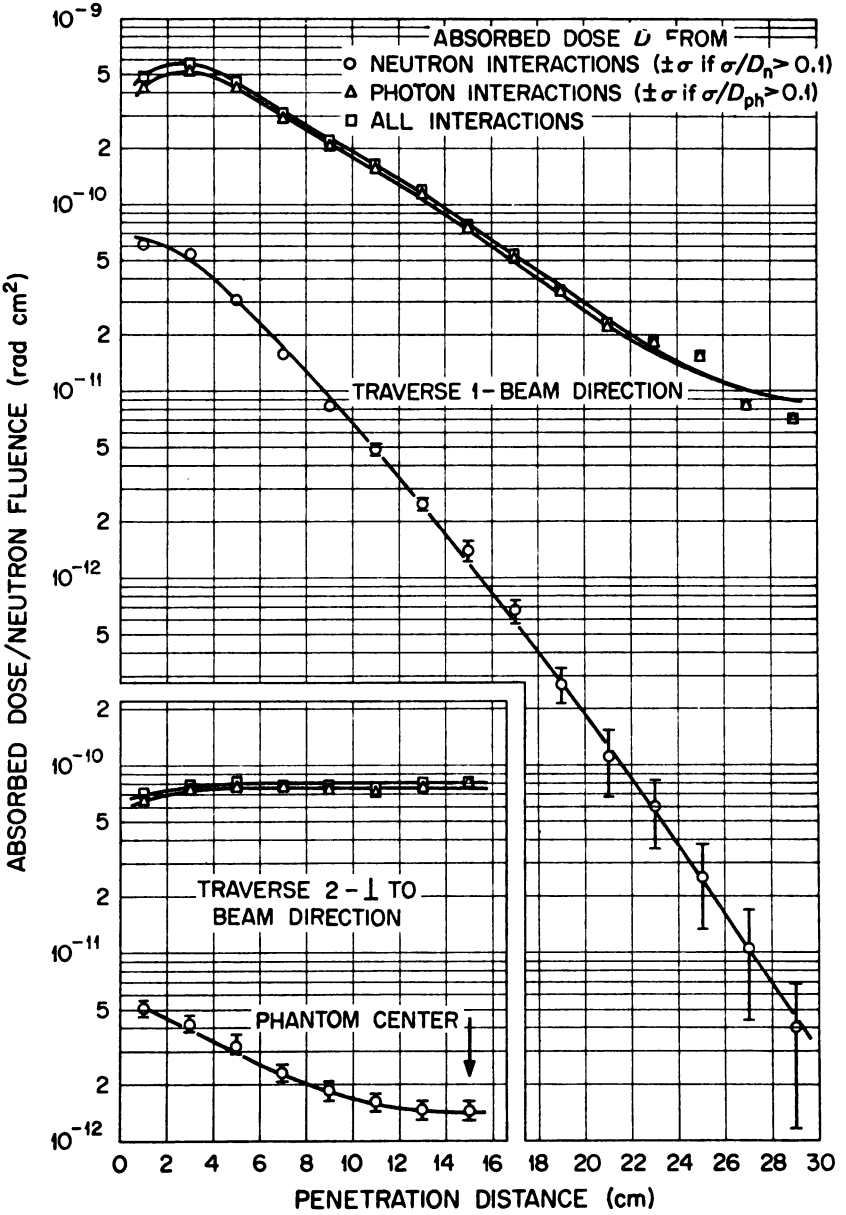

Fig. 31. Absorbed dose in central region of the cylindrical phantom. (Neutron energy 0.1 eV.)

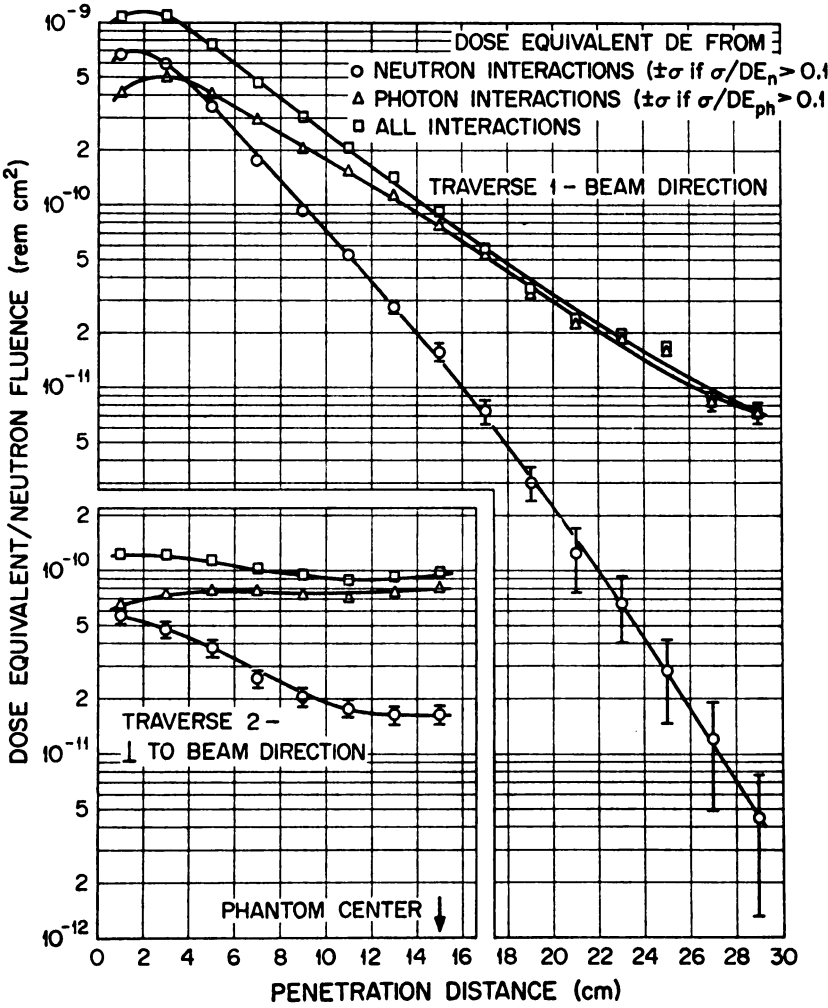

Fig. 32. Dose equivalent in central region of the cylindrical phantom. (Neutron energy 0.1 eV.)

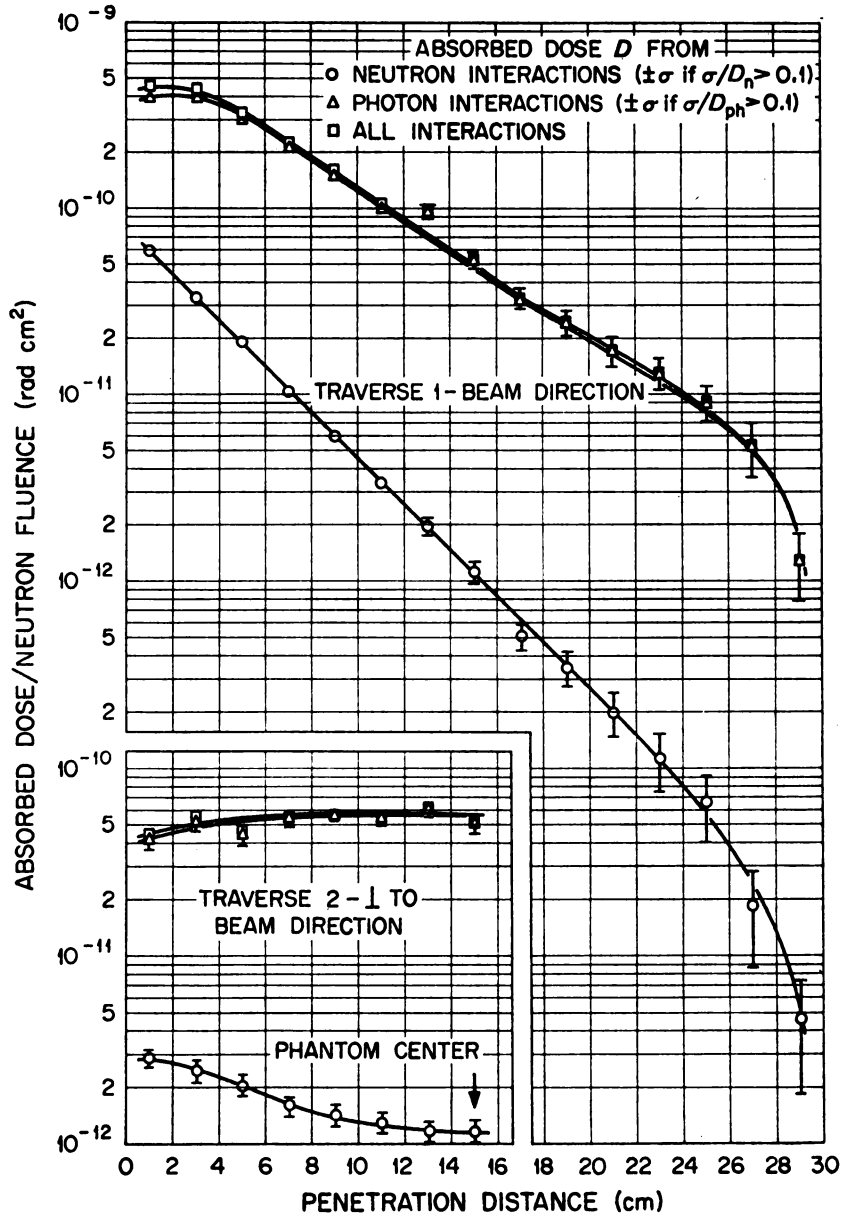

Fig. 33. Absorbed dose in central region of the cylindrical phantom. (Neutron energy 0.025 eV.)

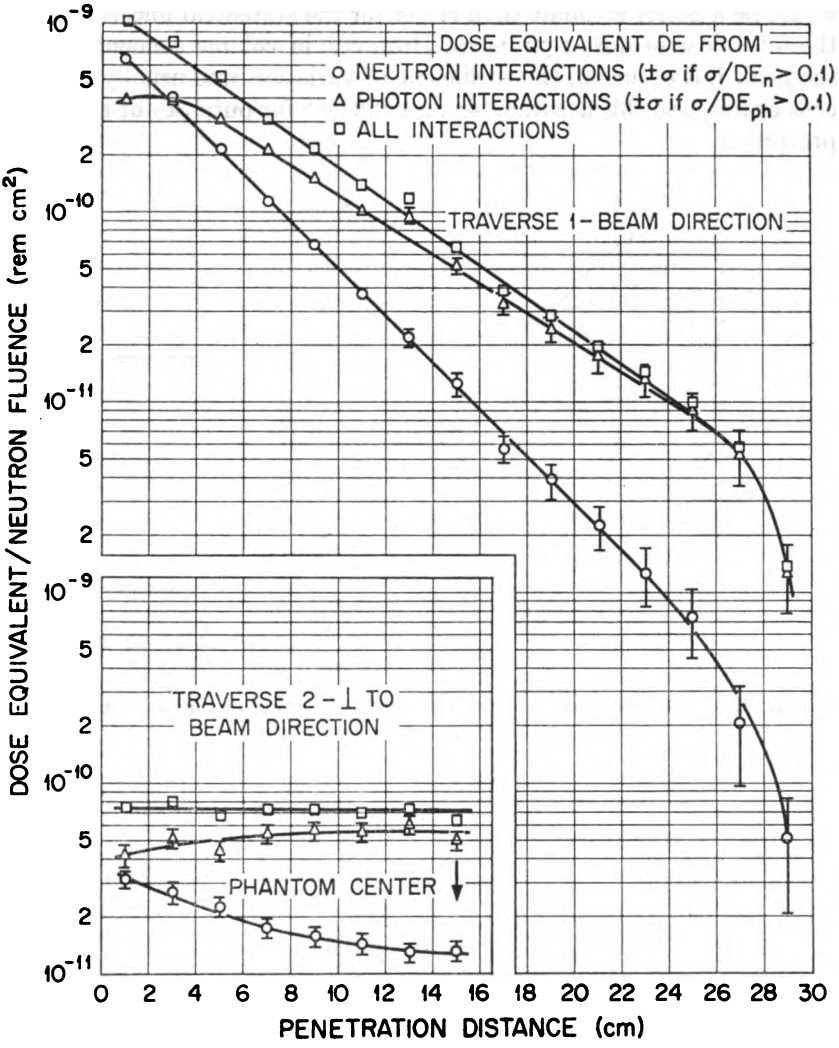

Fig. 34. Dose equivalent in central region of the cylindrical phantom. (Neutron energy 0.025 eV.)

variability. Perhaps a weighted combination of the standard deviations of the data for the photon interactions and for the neutron interactions would be a better estimate than either for the statistical uncertainty of the total dose or dose equivalent. However, in volume elements where the peak DE occurred, the coefficient of variation was usually 10 percent or less, and this accuracy seems sufficient for purposes of radiation protection.

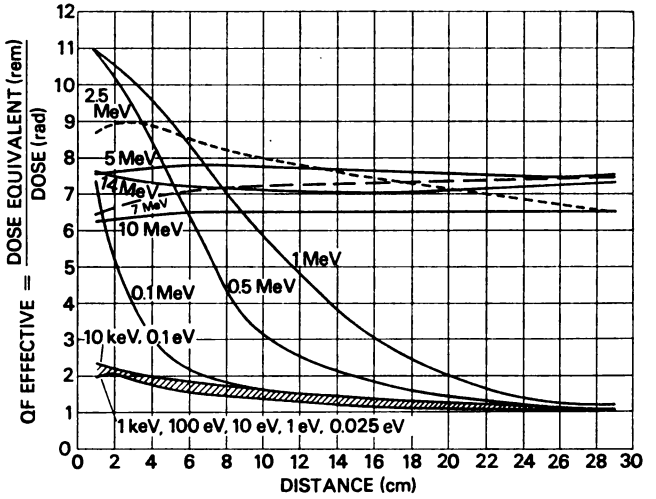

Fig. 35. Effective quality factor, $\overline{QF}$, in the direction of the neutron beam.

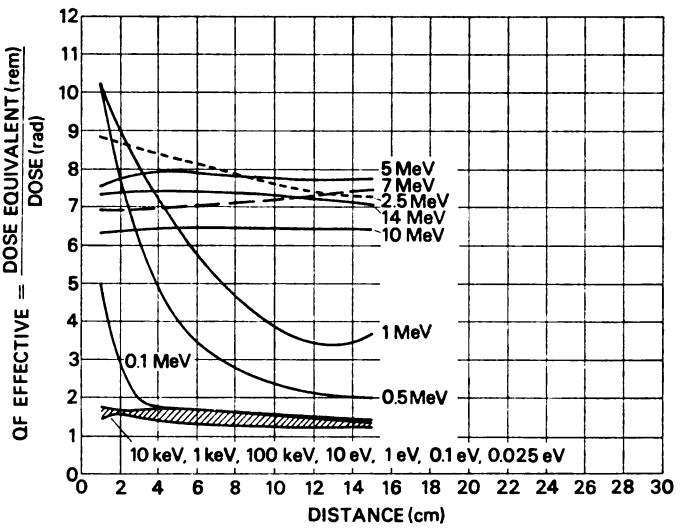

Fig. 36. Effective quality factor, $\overline{QF}$, along the lateral traverse.

Results for the volume elements adjacent to the bases of the cylinder are not reported in detail here. In general, the dose in these volume elements due to charged particles differed by less than 10 percent for neutron energies of 1 MeV and above, but were as much as 30–40 percent low at lower energies. The dose due to photons in these volume elements tended to be lower than for the central elements by 30–40 percent for all neutron energies.

One may define an effective quality factor, $\overline{QF}$, as the quotient DE/D at each of the computed points. $\overline{QF}$ varies from point to point because the energy spectrum of the radiation field varies within the phantom. $\overline{QF}$ has been computed for each volume element centered on Traverses 1 and 2 and smooth curves drawn through the computed points. These curves are shown graphically in Figures 35 and 36. The shaded areas contain a number of the curves, and it did not seem worthwhile to separate them in view of the limited over-all accuracy of the calculation.

B.II. Absorbed Dose and Dose Equivalent in Tissue Slabs for Neutron Energies from 0.5 to 400 MeV[9]

Theoretical dose-vs-depth curves have been obtained by Irving et al. (1967) for neutrons in the energy range 0.5 to 60 MeV and by Zerby and Kinney (1965) for neutrons in the energy range 60 to 400 MeV. All of these calculations were carried out for the case of an infinite beam of neutrons incident either normally or isotropically on one face of a semi-infinite slab of tissue 30 cm thick.

The calculations of Irving et al. were carried out using Monte Carlo methods and the neutron transport code 05R (Irving et al., 1965). The total cross sections for elastic and nonelastic nuclear scattering were taken insofar as possible from experimental data and were supplemented where necessary by optical-model calculations.[10] The angular distribution of elastically scattered neutrons was assumed to be anisotropic in such a manner that there was a linear variation of the cosine of the center-of-mass scattering angle. The energy distributions of the particles from nonelastic nuclear collisions were obtained using evaporation theory. A version of the evaporation code, written by Dresner (1962), which allows for the emission of neutrons, protons, and alpha particles, was used. In the calculations, charged particles were assumed to deposit their energy at their point of origin. Neutrons were transported until they reached a cut-off energy of 1 eV. When they reached this energy, they were displaced in a random direction a distance equal to the root-mean-square distance traveled from thermalization to absorption before inducing, with appropriate probability, an $H(n,\gamma)$ or an $N(n,p)$ reaction. Since the exact spectrum of γ rays following a nonelastic collision is not known, it was assumed that the excitation energy went entirely into 2.2 MeV γ rays. The absorbed dose from γ rays was obtained by means of an infinite medium build-up factor and a flux-to-absorbed dose conversion factor (Henderson, 1959). This very approximate treatment of the γ rays is justified by the fact that the γ-ray contribution to the total dose is small except at the very low energies.

The calculations of Zerby and Kinney were carried out using Monte Carlo methods and nucleon transport code NTC (Kinney, 1964). This

[9] The NCRP is greatly indebted to R. G. Alsmiller, Jr., for this contribution.

[10] The master cross section tape, compiled by Irving for use in the 05R code, as well as references to all of the data used, may be obtained from the Radiation Shielding Information Center of the Oak Ridge National Laboratory (Irving et al., 1965).

code treats neutrons below 50 MeV by using the 05R code mentioned above so that when neutrons fall below this energy their absorbed dose and dose equivalent are obtained in essentially the same manner as in the calculations of Irving et al. Protons below 50 MeV are assumed to slow down without nuclear interaction. Elastic scattering of nucleons above 50 MeV with protons is treated using experimental data,[11] and elastic scattering of nucleons above 50 MeV with heavier nuclei is neglected. The cross sections of nonelastic nuclear collisions above 50 MeV, as well as the energy and angular distributions of nucleons from such collisions, are taken from the intranuclear-cascade calculations of Bertini (1963 and 1965). In the calculations, the proton stopping power was obtained using the Bethe-Bloch formula (Bethe and Ashkin, 1964) and heavier charged particles were assumed to deposit their energy at their point of origin. The contribution of γ rays was neglected.

In both the Irving and the Zerby-Kinney calculations, the quality factor as a function of LET, used to convert from energy deposition per gram of tissue to dose equivalent, was taken from the tabular data given in NCRP Report No. 17 (NCRP 1954) and agrees closely with recommendations of the RBE committee of the ICRP and the ICRU (ICRP/ICRU, 1963). A constant value of 20 was assumed to apply to all LET values above 175 keV/μm. In the Zerby-Kinney calculations, it was arbitrarily assumed that a constant quality factor of 20 could be applied to all charged particles with mass heavier than that of a proton since their LET is usually above 175 keV/μm. In the Irving calculations, the quality factor was considered to be a function of LET for protons and alpha particles and was taken to be 20 for all particles with mass greater than that of an alpha particle.

Examples of absorbed dose per unit fluence vs. depth and dose equivalent per unit fluence vs. depth from the Irving et al. calculations are shown in Figures 37 through 44. In Figures 37 and 39, the various contributions to the absorbed dose as well as the total absorbed dose are shown as a function of depth for the case of a unit fluence of 30 MeV neutrons incident normally and isotropically, respectively, on one face of the slab. In Figures 38 and 40 various contributions to the dose equivalent, as well as the total dose equivalent, are shown as a function of depth for the case of a unit fluence of 30 MeV neutrons incident normally and isotropically, respectively, on one face of the slab. In Figures 41 through 44, analogous curves are shown for the case of 60 MeV incident neutrons. Similar results for all the energies considered by Irving et al. may be found in the cited reference (Irving et al., 1967).

[11] These data are the same as those used in the intranuclear-cascade calculations of Bertini (1963).

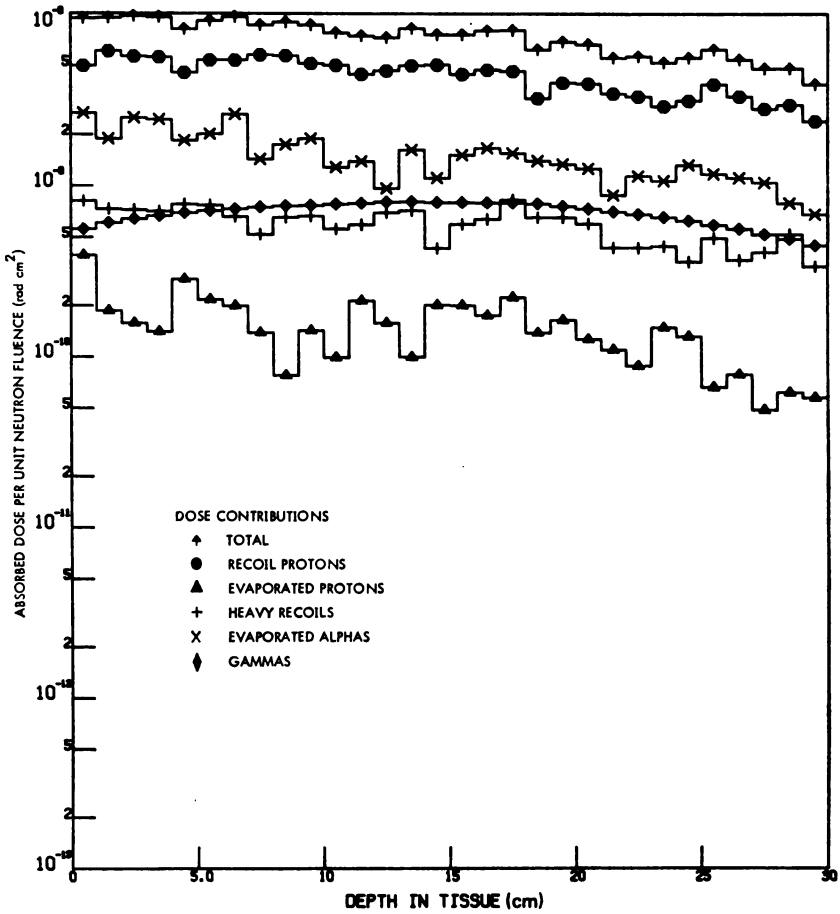

Fig. 37. Absorbed dose per unit fluence vs. depth in tissue for a unit fluence of 30 MeV neutrons incident normally on one face of a semi-infinite slab of tissue 30 cm thick.

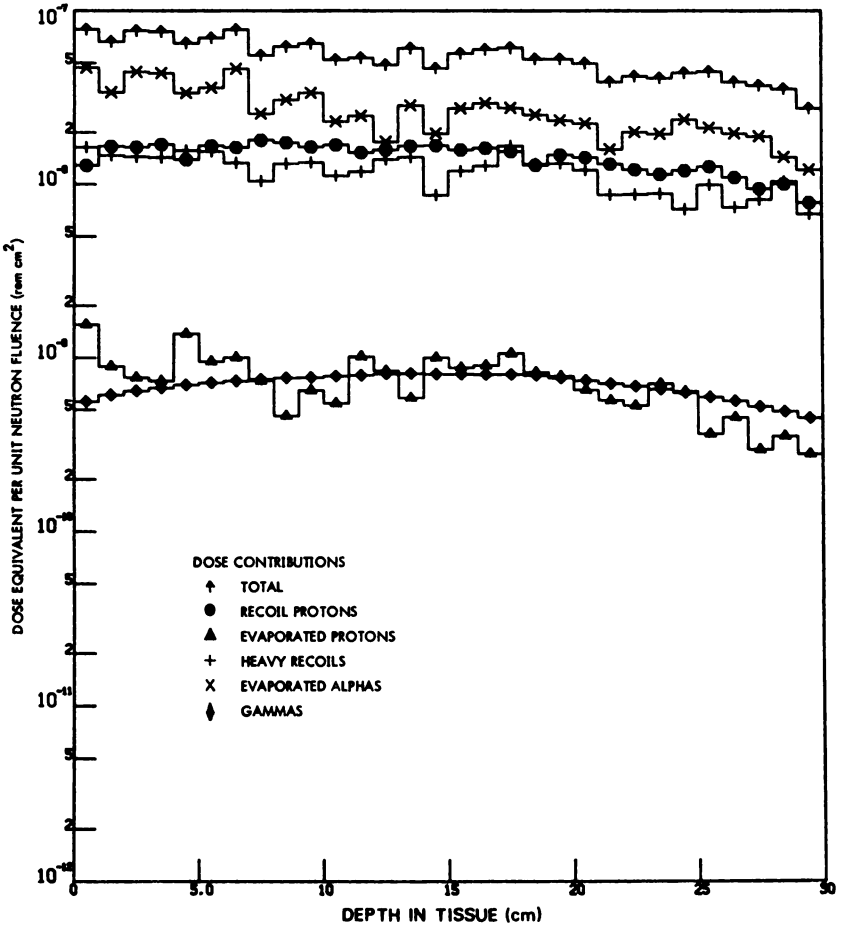

Fig. 38. Dose equivalent per unit fluence vs. depth in tissue for a unit fluence of 30 MeV neutrons incident normally on one face of a semi-infinite slab of tissue 30 cm thick.

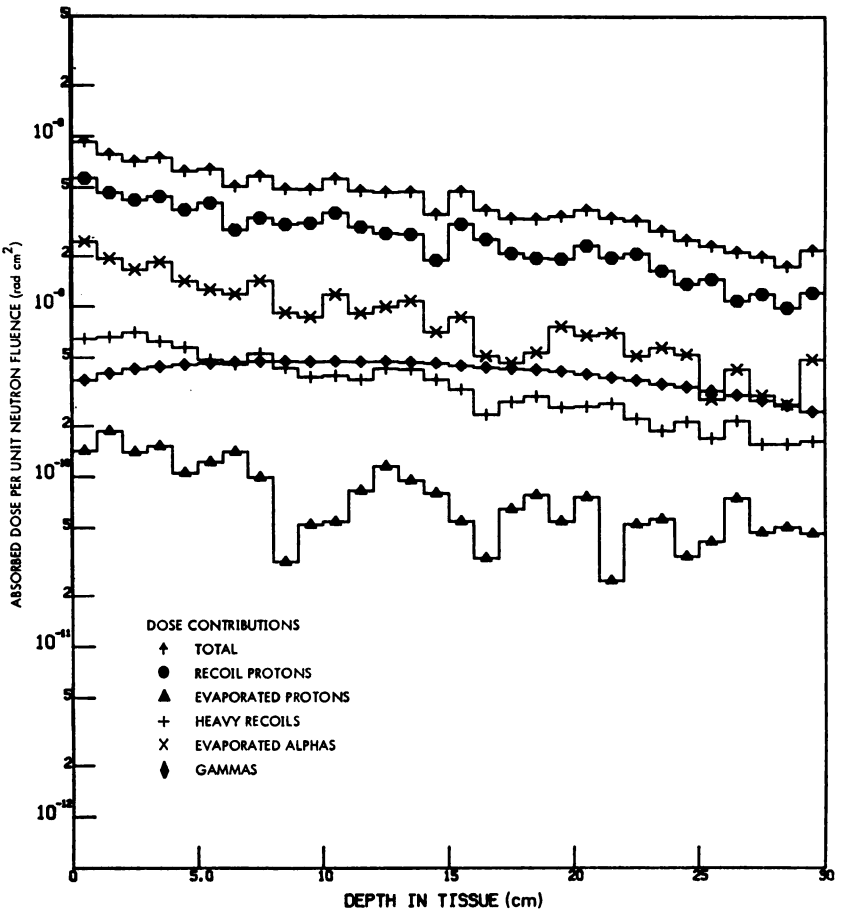

Fig. 39. Absorbed dose per unit fluence vs. depth in tissue for a unit fluence of 30 MeV neutrons incident isotropically on one face of a semi-infinite slab of tissue 30 cm thick.

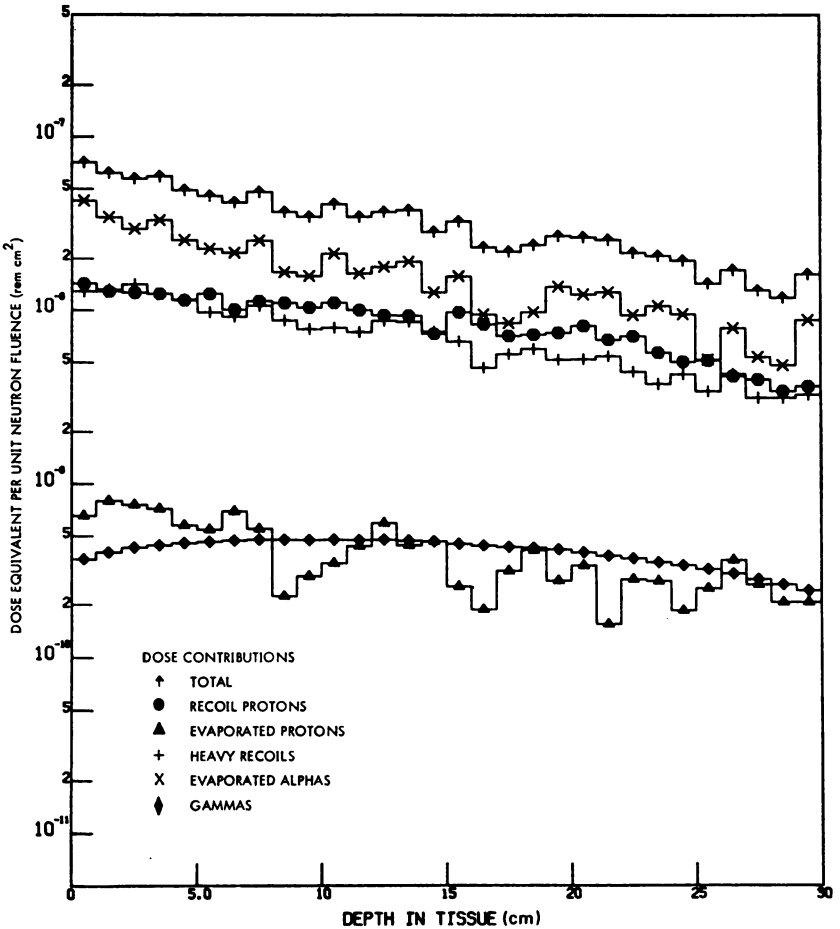

Fig. 40. Dose equivalent per unit fluence vs. depth in tissue for a unit fluence of 30 MeV neutrons incident isotropically on one face of a semi-infinite slab of tissue 30 cm thick.

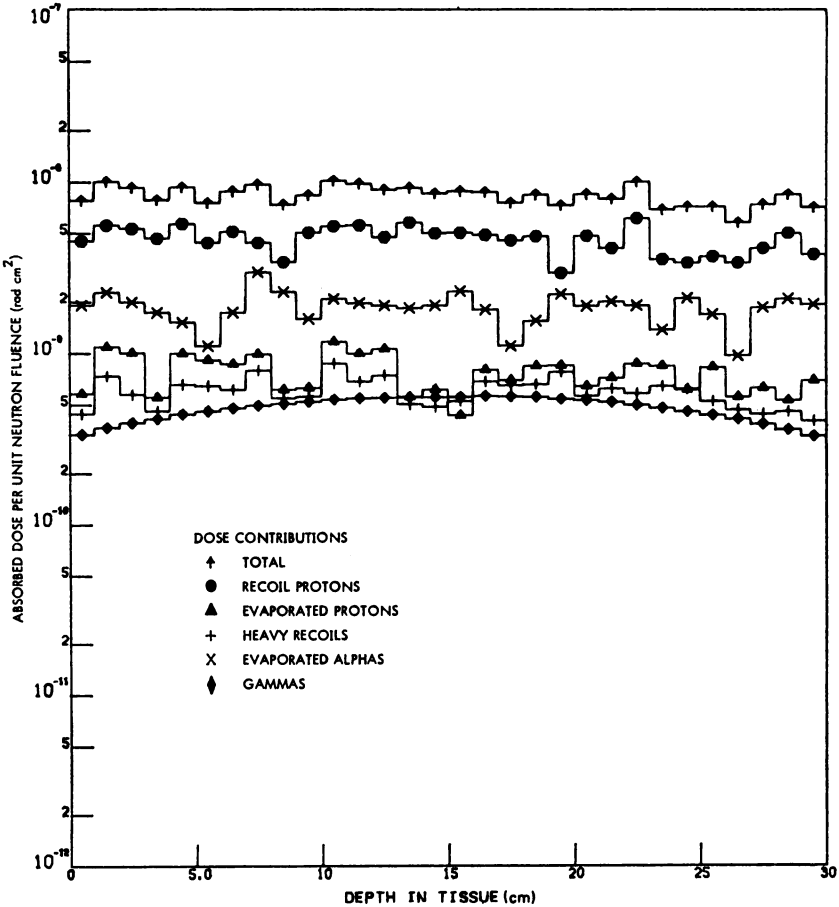

Fig. 41. Absorbed dose per unit fluence vs. depth in tissue for a unit fluence of 60 MeV neutrons incident normally on one face of a semi-infinite slab of tissue 30 cm thick.

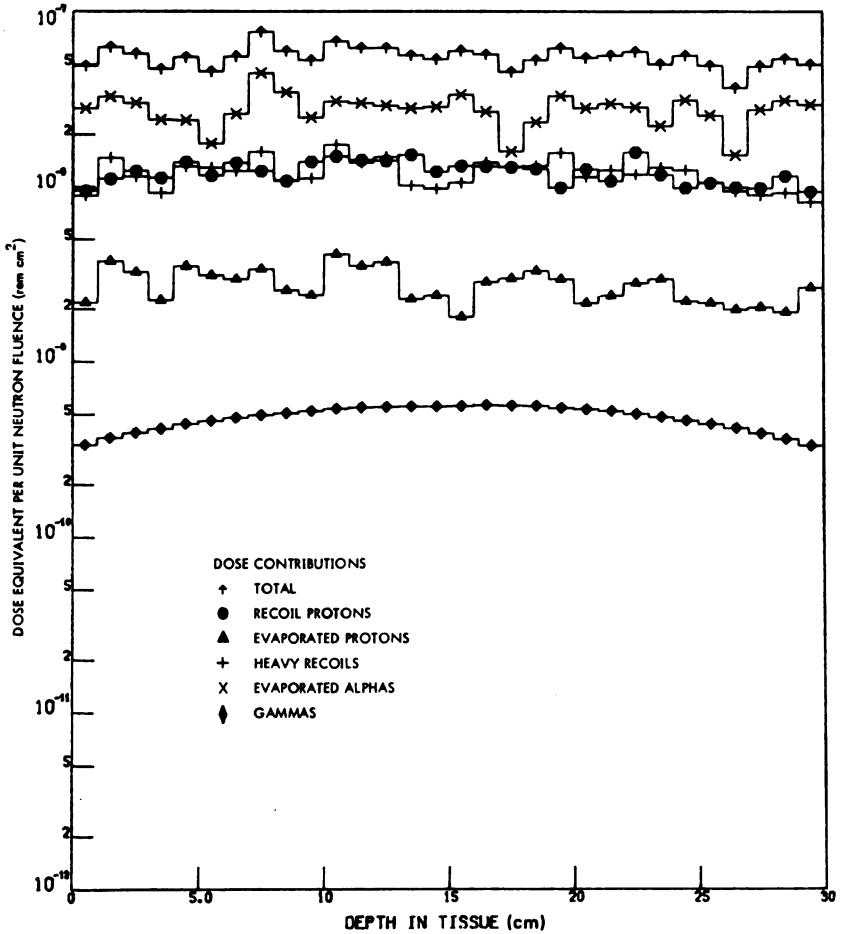

Fig. 42. Dose equivalent per unit fluence vs. depth in tissue for a unit fluence of 60 MeV neutrons incident normally on one face of a semi-infinite slab of tissue 30 cm thick.

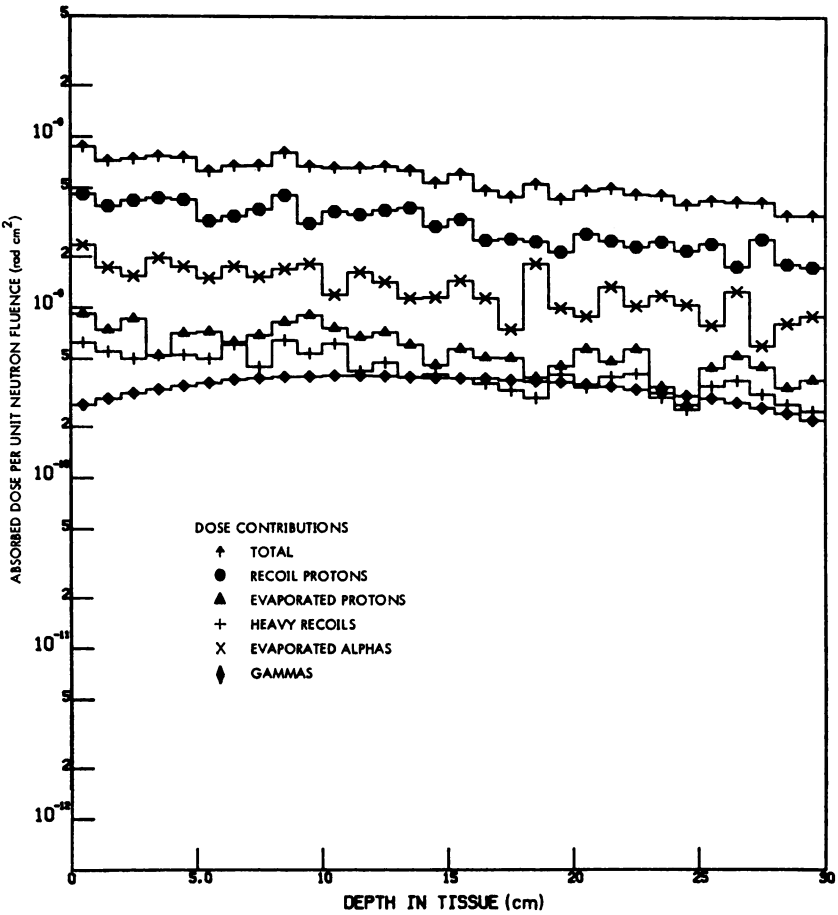

Fig. 43. Absorbed dose per unit fluence vs. depth in tissue for a unit fluence of 60 MeV neutrons incident isotropically on one face of a semi-infinite slab of tissue 30 cm thick.

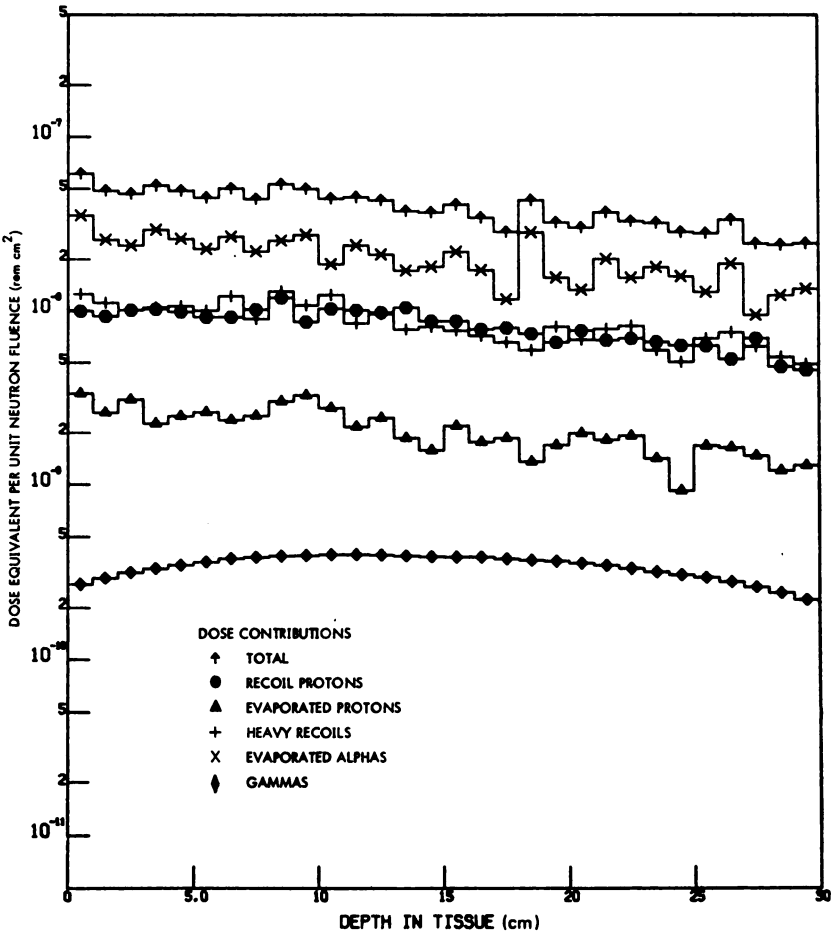

Fig. 44. Dose equivalent per unit fluence vs. depth in tissue for a unit fluence of 60 MeV neutrons incident isotropically on one face of a semi-infinite slab of tissue 30 cm thick.

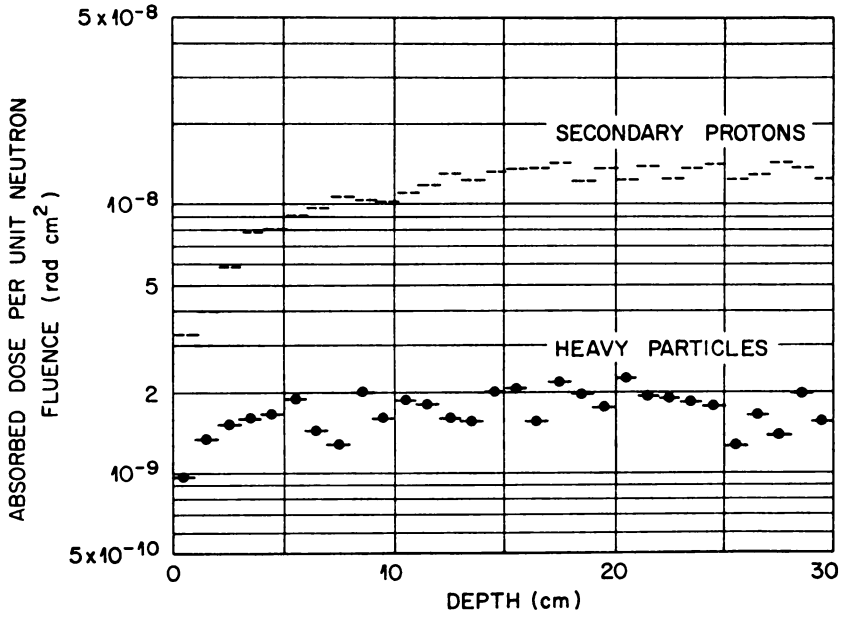

Fig. 45. Absorbed dose per unit fluence vs. depth in tissue for a unit fluence of 200 MeV neutrons incident normally on one face of a semi-infinite slab of tissue 30 cm thick. (From Zerbey, C. D. and Kinney, W. E., "Calculated Tissue Current-to-Dose Conversion Factors for Nucleons Below 400 MeV," Nucl. Inst. Methods **36**, 125, 1965.) [Reprinted with the permission of North-Holland Publishing Company, Amsterdam.]

In Figures 45 and 46 are shown the contributions to the absorbed dose as a function of depth for the case of a unit fluence of 200 MeV neutrons incident normally and isotropically, respectively, on one face of a slab of tissue 30 cm thick. Similar curves showing the dose equivalent as a function of depth in tissue are not given by Zerby-Kinney and therefore are not available, but a considerable amount of information on the dose equivalent may be found in the published paper (Zerby and Kinney, 1965).

The maximum dose equivalent for a unit incident-neutron fluence at any depth in the tissue is shown in Figure 47 as a function of neutron energy. The plotted points show the results of the calculations.

Several features of the results are worthy of note. The abrupt increase in the dose equivalent just above 10 MeV arises because, in the nuclear model used, alpha-particle emission begins to become an important process just above 10 MeV. The dose equivalent decreases in going from 30 to 60 MeV (Irving calculations) because the alpha par-

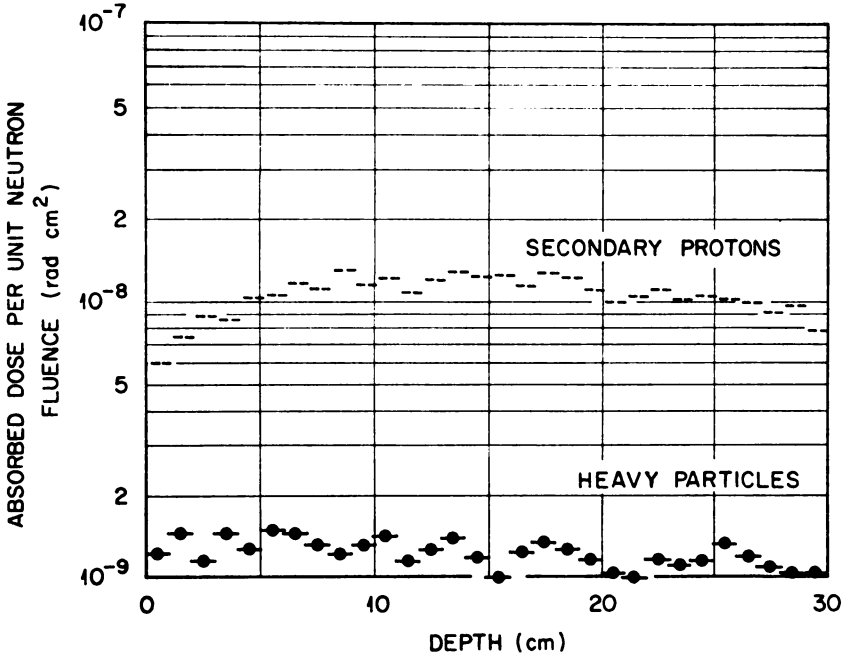

Fig. 46. Absorbed dose per unit fluence vs. depth in tissue for a unit fluence of 200 MeV neutrons incident isotropically on one face of a semi-infinite slab of tissue 30 cm thick. (From Zerby, C. D. and Kinney, W. E., "Calculated Tissue Current-to-Dose Conversion Factors for Nucleons Below 400 MeV," Nucl. Inst. Methods **36**, 125, 1965.) [Reprinted with the permission of North-Holland Publishing Company, Amsterdam.]

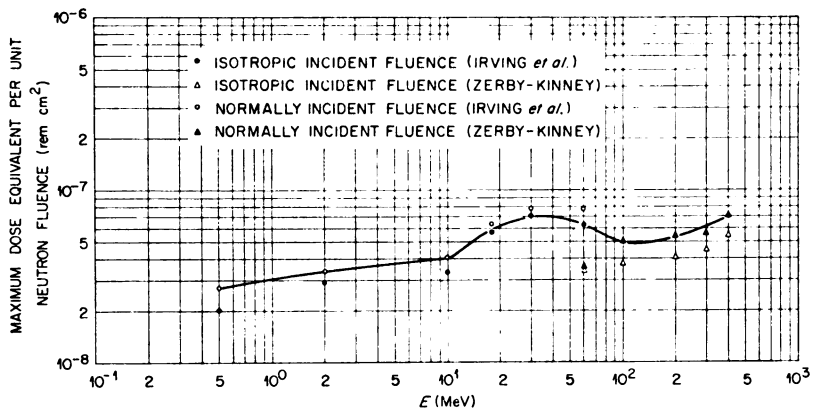

Fig. 47. Maximum dose equivalent for a unit incident-neutron fluence at any depth in tissue as a function of neutron energy.

ticles emitted from the higher energy neutron collisions have higher energy and thus a smaller quality factor.

The most notable characteristic of the results in Figure 47 is the discontinuity at $E = 60$ MeV. The calculations of Irving et al. and those of Zerby and Kinney do not agree at 60 MeV because of the different models used to describe production from neutron-nucleus collisions in the 50- to 60-MeV energy range. The discrepancy arises primarily because the evaporation theory used by Irving gives more alpha particles than does the cascade theory used by Zerby and Kinney. This discon-

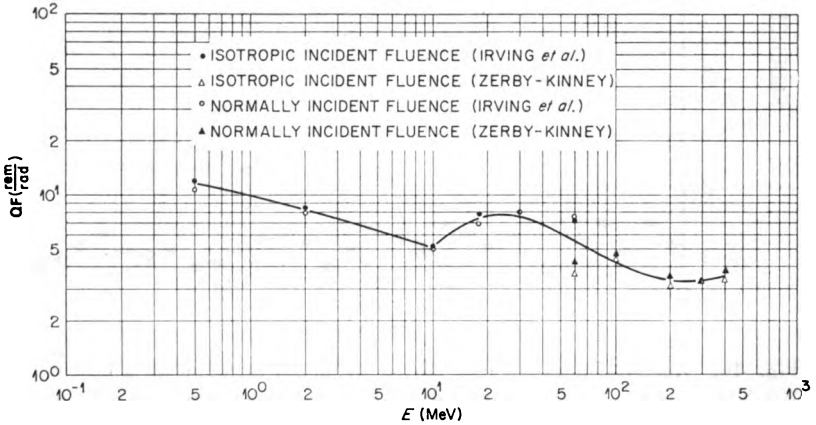

Fig. 48. $\overline{QF}$ (obtained by dividing the maximum dose equivalent at any depth in tissue by the maximum absorbed dose at any depth in tissue) as a function of neutron energy.

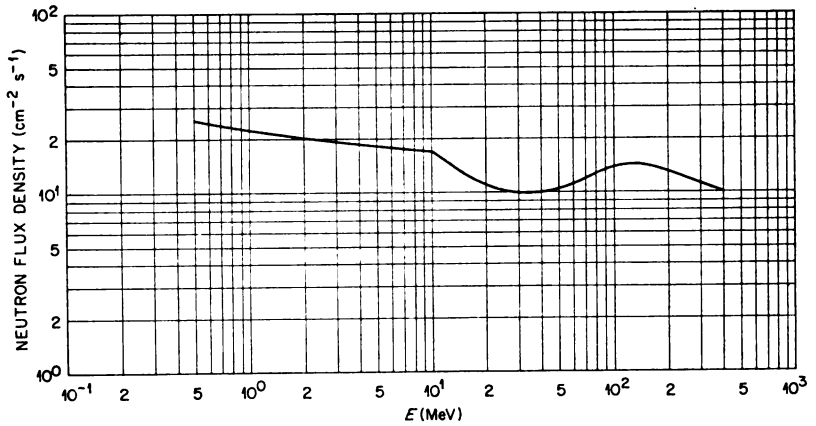

Fig. 49. Flux density of normally incident neutrons which gives a maximum dose rate of 0.1 rem per 40 hours as a function of neutron energy.

tinuity is undesirable, but unfortunately the experimental information needed to resolve it, that is, the energy and angular distributions of neutrons, protons, and alpha particles emitted when a 60 MeV neutron collides with light elements such as carbon, nitrogen, and oxygen, is not available.

The solid curve in Figure 47 has been drawn by eye to more or less average the values for the normally incident fluence in the vicinity of 60 MeV. The fact that the curve is biased toward the Irving calculations at 60 MeV is not intended to imply that the Irving calculations are thought to be more valid, but is rather an attempt to be conservative. A similar curve could, of course, be drawn through the point for an isotropically incident fluence, but it is not shown to emphasize the fact that the maximum permissible flux density (see Figure 49) is derived from the solid curve that is given.

In Figure 48, $\overline{QF}$, the average quality factor, obtained by dividing the maximum dose equivalent at any depth in the tissue by the maximum absorbed dose at any depth in the tissue, is given. Again the plotted points show the calculated results and the curve indicates a rough average of all the points.

In Figure 49 the flux density of normally incident neutrons which give a maximum dose rate of 0.1 rem/40 h is shown. This curve was obtained from the solid curve shown in Figure 47 and therefore implies the same rough average.

APPENDIX C

Reactions Employed in Neutron Production

Table 5 summarizes data on reactions usually used for the production of neutrons. Both cross section and thick target neutron yield are functions of the energy of the incident particle, which is given in MeV. At other incident particle energies, corresponding values may be quite different. In addition to the reactions listed, all elements will produce neutrons when bombarded with particles of energies greater than 20 MeV per nucleon. The number of neutrons produced per incident charged particle varies considerably with the material of the target and the incident charged particle up to an energy of about 100 MeV. Above 100 MeV, the number of neutrons produced per incident proton increases roughly linearly with the energy of the incident particle and increases approximately linearly with the atomic mass of the target nucleus. The information in the last two columns of Table 5 is qualitative only and is furnished as a rough guide to further information, especially in the variation of the cross sections with energy and angle. (See Kim, Milner and McGowan, 1967, and other references listed therein. See also Marion and Fowler, 1960.)

TABLE 5a—*Data on neutron production*

Target	Residual nucleus	Q/MeV	Reaction yields: Thick target yield (n/μC) or nominal cross section (barns)	Corresponding incident particle energy (MeV)	Degree of gamma-ray contamination	Gamma-ray energies (MeV)	Residual activity or comparative dosage
			(p,n) Reactions				
^{3}H	^{3}He	−0.76	0.50 barn	3.0 (peak)	Considerable $\simeq 10^{-8}$ γ/proton	20	None
^{7}Li	^{7}Be	−1.65	0.50 barn	2.2 (peak)	Considerable $\simeq 2 \times 10^{-8}$ γ/proton	17	Considerable (^{7}Be)
^{9}Be	^{9}B	−1.85			Considerable $\simeq 10^{-9}$ γ/proton		None
			(d,n) Reactions				
^{2}H	^{3}He	3.26	2.5×10^{6} n/μC	0.25	None	None	None
			0.05 barn	0.25			
^{3}H	^{4}He	17.6	~108 n/μC	0.20	Slight		None
			5 barn	0.10			
^{6}Li	^{7}Be	3.38			0.3 γ/n	0.4	Moderate (^{7}Be)
^{6}Li	$^{3}He + ^{4}He$	1.79	1.7×10^{7} n/C	0.6		0.4	None
^{7}Li	$^{6}He + ^{4}He$	15.1	0.06 barn	0.2	0.3 γ/n	4.9	None
^{9}Be	^{10}B	4.36	4×10^{6} n/μC	0.4		3.6; 2.2; 1.7; 1.0; 0.7; 0.4	Small (^{10}Be)
			0.4 barn	5.0			
^{10}B	^{11}C	6.47				8.9; 6.7; 6.4; 4.5	Small (^{11}C)
^{11}B	^{12}C	13.7					None
^{12}C	^{13}N	0.28	35×10^{-3} barn/ster (0°)	2.2–3	4.45 γ/n		Slight (^{13}N)

TABLE 5a—*Continued*

Target	Residual nucleus	Q/MeV	Reaction yields: Thick target yield (n/μC) or nominal cross section (barns)	Corresponding incident particle energy (MeV)	Degree of gamma-ray contamination	Gamma-ray energies (MeV)	Residual activity or comparative dosage
^{13}C	^{14}N	5.3			None	6.1; 5.0; 3.3; 2.3; 1.6; 0.7	None
^{14}N	^{15}O	5.12					Slight (^{15}O)
^{19}F	^{20}Ne	10.65				10.6; 9.1; 8.4; 6.4; 5.2; 3.5; 2.8; 1.6; 0.5	
			(γ,n) Reactions				
^{2}H	^{1}H	−2.226	2.7×10^{-8} barn	4.0 (max)			
^{9}Be	^{8}Be (2α)	−1.666	1×10^{-3} barn	1.7 (max)			

TABLE 5b—*Data on neutron production*

(α,n) Reactions

Target	Residual nucleus	Q/MeV	Yield ns^{-1} Ci^{-1}	Reaction	Degree of gamma-ray contamination	Gamma-ray energies (MeV)	Residual activity or comparative dosage
^{7}Li	^{10}B	−2.78	5×10^{4}	Po-Bi		0.8; 0.48	$n/\gamma \simeq 0.25$
^{9}Be	^{12}C		2.3×10^{6}	Po-Be		0.8; 4.45	$n/\gamma \simeq 4.0$
^{9}Be	^{12}C	5.7	1.7×10^{7}	Ra-Be	Considerable		$n/\gamma < 0.01$
^{10}B	^{13}N	1.07	6×10^{5}	Po-B		0.8; 3.7; 2.4	$n/\gamma \simeq 5.0$
^{11}B	^{14}N	0.15	7×10^{6}	Ra-B	Considerable		$n/\gamma < 0.01$
			2×10^{5}	Po-F		0.8; 1.3; 1.5	$n/\gamma \simeq 1.0$
^{19}F	^{22}Na	−2.3	2.5×10^{6}	Ra-BeF_4	Considerable		$n/\gamma < 0.01$
^{23}Na	^{26}Al	−3.88	4×10^{4}	Po-Na		0.8; 0.4; 1.1; 1.8; 2.6	$n/\gamma \simeq 0.03$
^{25}Mg	^{28}Si	2.67	3×10^{4}	Po-Mg		0.8; 1.8; 4.0	$n/\gamma \simeq 0.15$
^{26}Mg	^{29}Si	0.04					
^{27}Al	^{30}P	−2.92	2×10^{4}	Po-Al		0.8; 1.3; 2.3; 3.6	$n/\gamma \simeq 0.1$

APPENDIX D

Neutron Capture Gamma Rays

Table 6 summarizes neutron capture cross sections and maximum gamma-ray energies encountered in neutron capture. The gamma-ray spectra usually consist of a number of individual lines where the lines are well separated. At lower energies, where the lines are very close and cannot be separated, the spectra appear to be continuous. The last column in Table 6 gives the average number of photons per capture, where they have been measured. For more complete information on the energies of the gamma rays and the intensities of lines, see Bartholomew et al. (1967) and Groshev et al .(1968 and 1969).

TABLE 6—*Capture gamma rays*[a]

Target	Thermal (n,γ) Cross Section	Highest Energy Gamma Ray	Average Number of Photons per Capture
	Barns	MeV	
Aluminum	0.235	7.724	2
Antimony	5.7	6.80	
Arsenic	4.3	7.30	
Barium	1.2	9.23	2.7
Beryllium	0.0095	6.814	
Bismuth	0.034	4.17	
Boron-10	3837	0.478[b]	0.94
Cadmium	2450	9.046	4.1
Calcium	0.44	7.83	
Carbon-12	0.0034	4.95	1.3
Chlorine	33.2	8.56	3.1
Chromium	3.1	9.716	2
Cobalt	37.0	7.486	
Copper	3.77	7.914	2.6
Fluorine	0.008	6.600	
Gadolinium	46000	7.78	3.9
Gold	98.8	6.494	3.5
Hydrogen-1	0.332	2.230	1.0
Hydrogen-2	0.00057	6.251	1.0
Indium	196	6.38	3.3
Iron	2.53	10.16	1.7
Lead	0.17	7.38	
Lithium	70.7	7.26	
Magnesium	0.063	11.089	
Manganese	13.2	7.261	2.6
Mercury	380	7.66	3.3
Molybdenum	2.7	9.15	
Nickel	4.8	8.997	
Niobium	1.15	7.19	2.6
Nitrogen-14	0.075	10.833	
Phosphorus	0.190	7.94	
Platinum	8.8	7.920	
Potassium	2.1	9.39	
Praseodymium	11.6	5.83	
Rhodium	156	6.792	
Samarium	5600	7.89	5.6
Scandium	24	8.85	
Selenium	12.3	10.483	
Silicon	0.160	10.599	
Silver	63	7.27	2.9

TABLE 6—*Continued*

Target	Thermal (n,γ) Cross Section	Highest Energy Gamma Ray	Average Number of Photons per Capture
	Barns	Mev	
Sodium	0.534	6.41	2
Strontium	1.21	9.22	
Sulfur	0.52	8.64	
Tantalum	21	6.07	
Thallium	3.4	6.54	
Tin	0.625	9.35	
Titanium	5.8	10.621	
Tungsten	19.2	7.42	
Vanadium	4.98	7.98	2.5
Zinc	1.10	9.51	
Zirconium	0.18	8.66	

[a] These data are taken from Mittelman and Liedtke (1955), BNL (1958), Bartholomew, et al. (1967), Groshev, et al. (1968 and 1969); and BNL (1964).

[b] Gamma ray from excited state of ^{7}Li following (n,α) reaction.

APPENDIX E

Shielding Data[12]

In recent years shielding theory has benefited by a substantial growth of the body of input information and of computing methodology. Some of these developments are utilized in this appendix which deals with the neutron shielding properties of ordinary concrete and of several other useful shielding materials; it also includes a review of the removal cross section method and a few remarks on the availability of other neutron shielding calculation techniques.

Ordinary Concrete

The importance of concrete as a structural and as a shielding material makes it especially worthwhile to have detailed analysis of its shielding characteristics. Figures 50 through 59 present results of Monte Carlo calculations of the penetration through concrete of normally incident neutrons at various (monoenergetic) source energies (Clark et al., 1966 and Schmidt, 1968). The source energies are 14, 12, 10, 8, 6, 4, 3, 2, 1.3, and 0.7 MeV. The concrete is taken to represent ordinary concrete, density 2.43 g/cm^3, with water content of 0.14 g/cm^3. For purposes of computation this concrete was represented by the following material concentrations:

Material	Atoms/cm^3 ($\times 10^{-24}$)
Hydrogen	0.0093
Oxygen	0.0463
Silicon	0.00992
Calcium	0.00883
Carbon	0.0065

The curves in these figures give the neutron kerma in tissue per unit fluence in units of ergs/g per square centimeter. The solid curve in each

[12] The NCRP is greatly indebted to F. H. Clark for this contribution.

figure gives the neutron kerma in tissue immediately behind an infinite slab of the thickness indicated on the abscissa. The dashed curve gives the neutron kerma in tissue in an infinite half space of concrete at the depth from the surface given by the abscissa value. In both cases the source is an infinitely broad beam of neutrons normally incident on the bounding face of the concrete.

The statistical errors (relative standard errors) associated with these curves range from less than 10 to 15 percent for calculated attenuations of up to 10^3–10^4, to 10 to 25 percent for calculated attenuations of 10^4–10^5, and up to 50 percent for calculated attenuations greater than 10^5. The statistical error associated with a value read from the curve can reasonably be expected to be less than that determined in the calculation of a nearby single point on account of the stabilizing influence that arises from fitting a curve through many points.

The curves in Figures 50 through 59 can be used to assess the neutron tissue kerma behind a concrete wall when a fairly broad neutron beam is incident almost normally from the other side. (For a narrow or a distinctly non-normally incident beam, the kerma will be less than that represented by these curves.) It should be noted, however, that the tissue kerma due to gamma radiations produced in the shield may often be

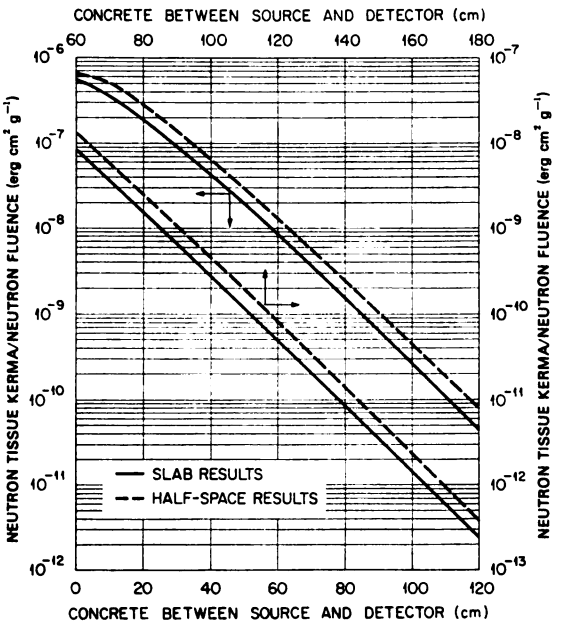

Fig. 50. Penetration of normally incident neutrons of energy 14 MeV through slabs and into half spaces of ordinary concrete.

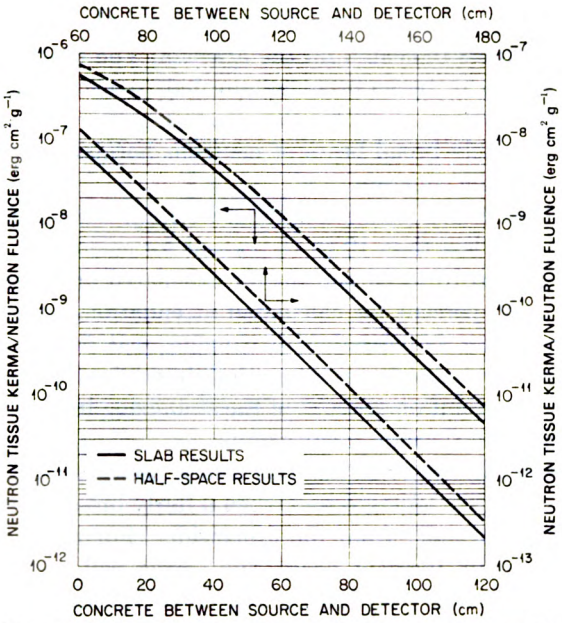

Fig. 51. Penetration of normally incident neutrons of energy 12 MeV through slabs and into half spaces of ordinary concrete.

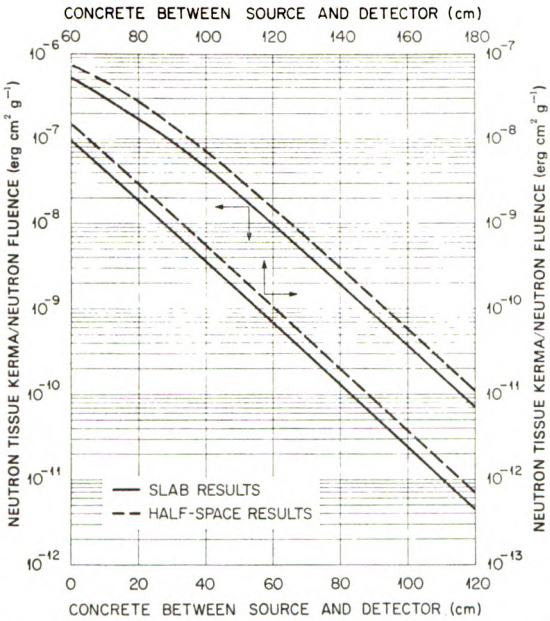

Fig. 52. Penetration of normally incident neutrons of energy 10 MeV through slabs and into half spaces of ordinary concrete.

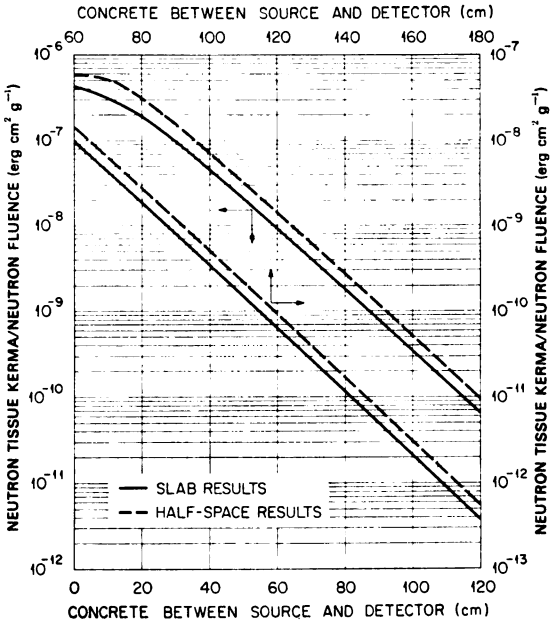

Fig. 53. Penetration of normally incident neutrons of energy 8 MeV through slabs and into half spaces of ordinary concrete.

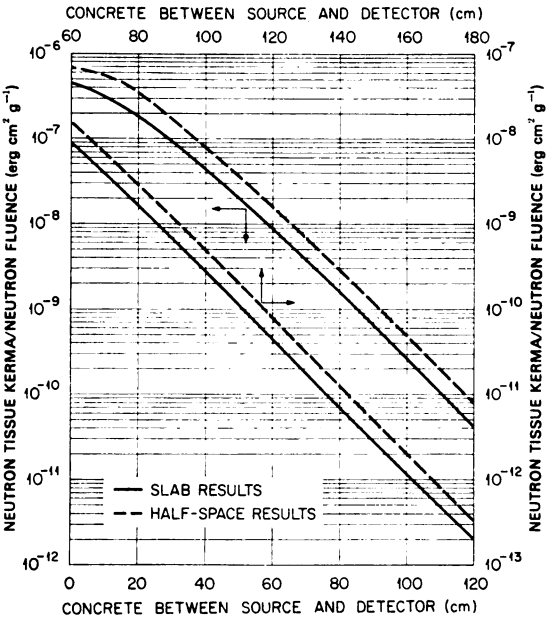

Fig. 54. Penetration of normally incident neutrons of energy 6 MeV through slabs and into half spaces of ordinary concrete.

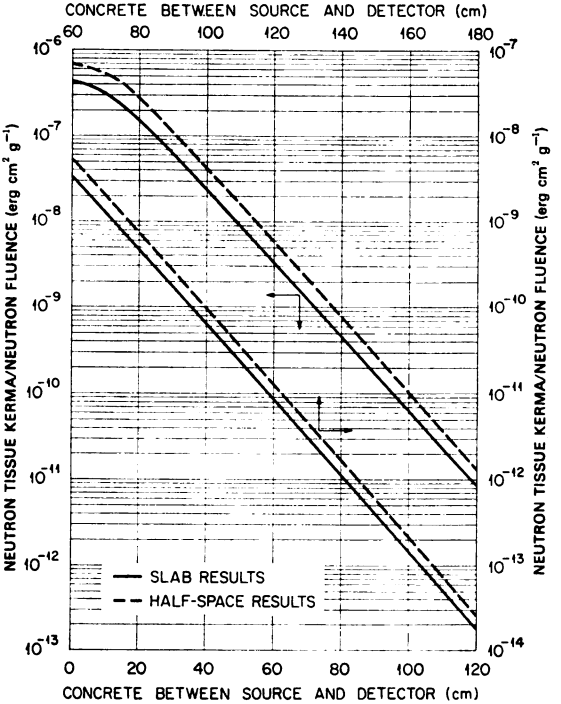

Fig. 55. Penetration of normally incident neutrons of energy 4 MeV through slabs and into half spaces of ordinary concrete.

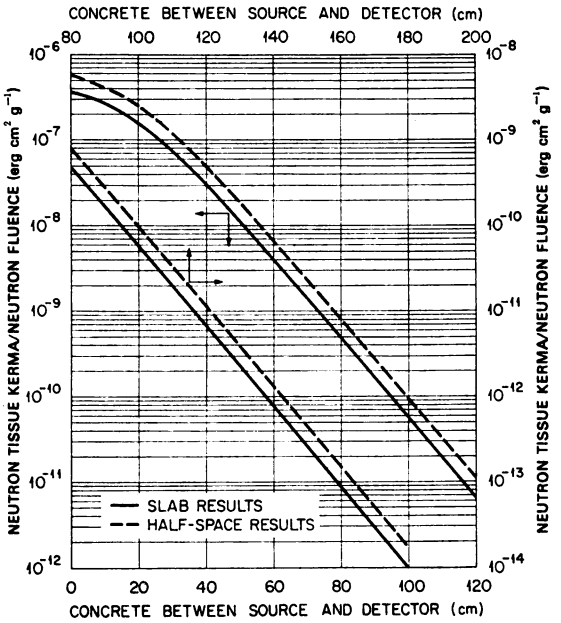

Fig. 56. Penetration of normally incident neutrons of energy 3 MeV through slabs and into half spaces of ordinary concrete.

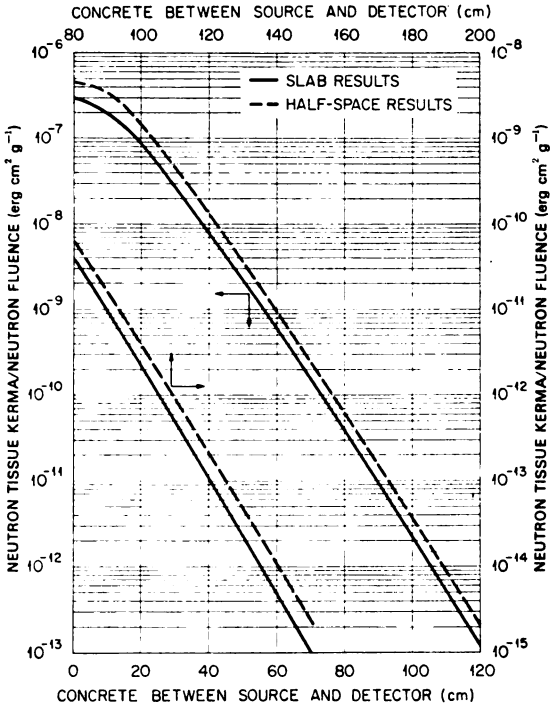

Fig. 57. Penetration of normally incident neutrons of energy 2 MeV through slabs and into half spaces of ordinary concrete.

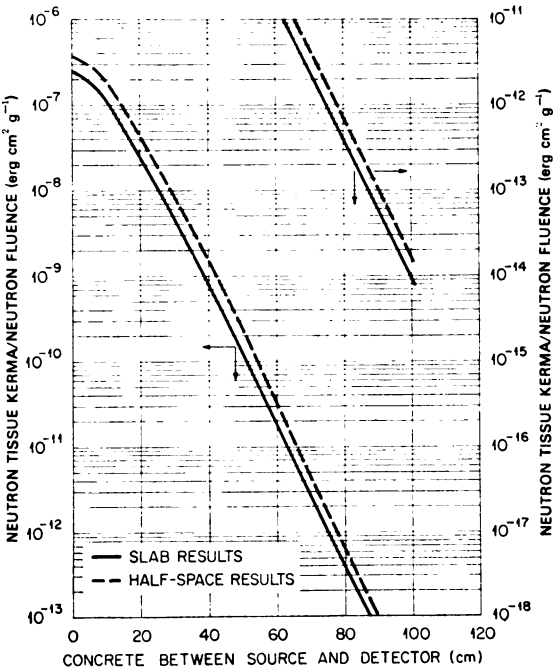

Fig. 58. Penetration of normally incident neutrons of energy 1.3 MeV through slabs and into half spaces of ordinary concrete.

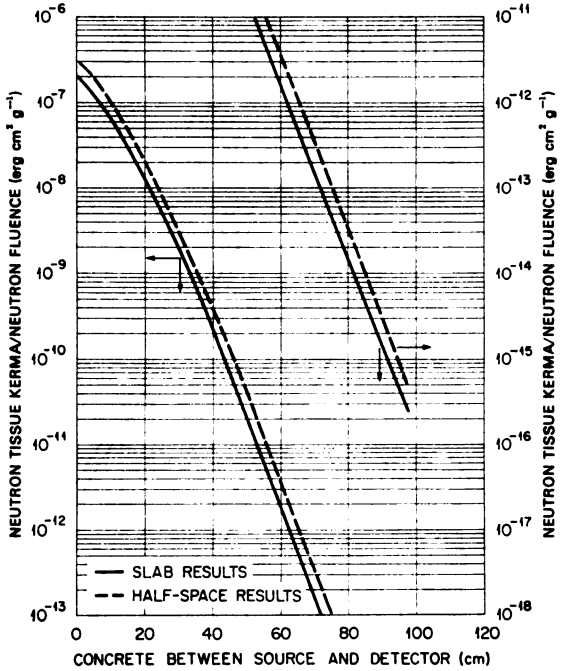

Fig. 59. Penetration of normally incident neutrons of energy 0.7 MeV through slabs and into half spaces of ordinary concrete.

more important than that due to the neutrons that have penetrated (see next section below).

In the case of irradiation by neutrons having a spectral distribution $\Phi(E)$ (the fluence of neutrons of energy E per unit interval of E), $K(t)$ the total neutron kerma in tissue at depth (t) can be obtained by appropriate integration. Thus if $\kappa(E, t)$ is the kerma in tissue at depth, t, due to neutrons of incident energy E,

$$K(t) = \int_0^\infty \kappa(E, t)\, \Phi(E)\, dE. \tag{E2}$$

This may be approximated by

$$K(t) = \sum_i \kappa(E_i, t)\, \Phi(E_i)\, \Delta E_i \tag{E3}$$

where the energy range over which $\Phi(E)$ is significant has been cut into a number of segments ΔE_i, each such segment containing one energy point E_i corresponding to a source energy of one of the Figures 50 to 59. $\kappa(E_i, t)$ is the value of the solid curve from source energy E_i corresponding to a wall thickness of t cm. $\Phi(E_i)$ is given by

$$\Phi(E_i)\Delta E_i = \int_{\Delta E_i} \Phi(E)\, dE. \qquad \text{(E4)}$$

The dashed curves in Figures 50 to 59 represent neutron penetration through a semi-infinite concrete medium. After one or two relaxation lengths of penetration, they should differ very little from the penetration curve of an infinite medium. Much of the existing body of data on the penetration of concrete by neutrons is for infinite media (Trubey and Emmet, 1965). Such data, where they appear otherwise appropriate to the needs of a particular situation, might be fairly well adapted to a finite system by correcting them in the proportion of the dashed-to-solid curve value at the proper penetration distance and energy.

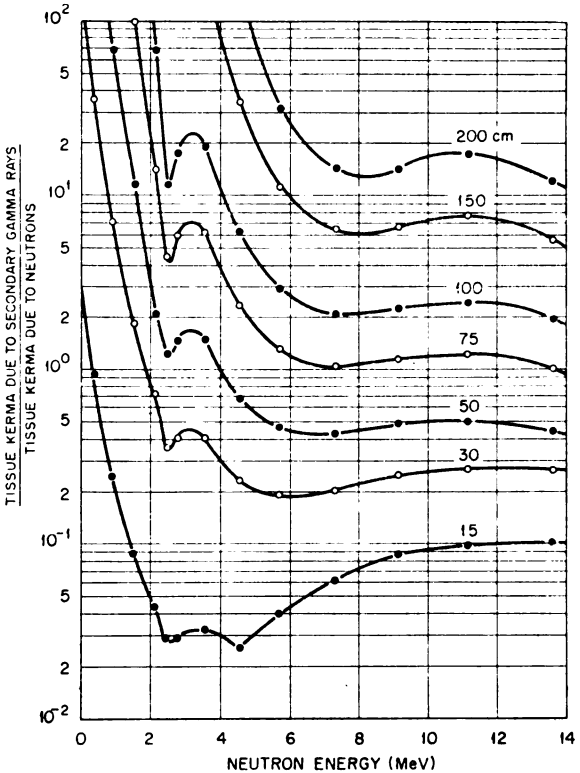

Fig. 60. Ratio of tissue kerma due to secondary gamma rays to tissue kerma due to neutrons, as a function of neutron energy. Curves are given for seven concrete slab thicknesses. The points are shown at the centers of contiguous neutron energy groups.

Gamma Radiation Produced by Shielding

The above results are valid within the limitations noted, but they do not include the effects of gamma radiation produced by interaction of the neutrons in the concrete. The secondary gamma rays may be more penetrating than the neutrons, especially for low neutron energies, and thus dominate the tissue kerma for greater thicknesses of concrete.

The secondary gamma rays arise mostly from the capture of thermal or nonthermal neutrons (see Appendix D), although neutron inelastic scattering also contributes. Lack of cross section information and problems in the transmission calculations have prevented more than approximate estimates of the effects of secondary gamma rays in the past. Recently, however, Schmidt (1968) has used the ANISN discrete ordinates code (Engle, 1967) for calculations of neutron transmission through concrete slabs. This code utilizes cross sections for gamma-ray production including both thermal and nonthermal capture and inelastic scattering, and it includes the subsequent gamma-ray transport. Schmidt's calculations were made for neutrons incident at eight angles with respect to the slab and for 22 energy groups. Calculations for gamma-ray transmission were also made for 18 energy groups.

The concrete composition considered by Schmidt differed from that given above for ordinary concrete. The composition used by Schmidt is shown below.

Material	Atoms/cm³ ($\times 10^{-24}$)
Hydrogen	0.0085
Oxygen	0.0355
Carbon	0.0202
Magnesium	0.00186
Aluminum	0.000556
Silicon	0.01070
Calcium	0.0111
Iron	0.000193
Sodium + Potassium	0.000057
(Density = 2.3 g/cm³)	

It is important to note that variations in concrete compositions will affect the secondary gamma-ray production much more than the neutron transmission. This effect arises mostly due to the nonthermal or resonance energy neutron capture.[13]

Of the above transmission data, those for neutrons nearly normally

[13] If this variation appears to be significant, the reader may wish to check for the latest information available at the Radiation Shielding Information Center (RSIC), Oak Ridge National Laboratory, P. O. Box X, Oak Ridge, Tennessee 37830.

incident (cos of angle from normal = 0.9894) are used here to illustrate the relative importance of secondary gamma rays. Figure 60 shows the ratio of tissue kerma due to gamma rays to that due to neutrons as a function of neutron energy. Curves are given for seven slab shield thicknesses.

The secondary gamma rays are less important for higher neutron energies. For the high-energy neutron transport in tissue, described in Appendix B.II, the secondary gamma rays from thermal-neutron capture have been taken into account. They are not very significant (except for the lowest neutron energies) for any of the rather thin shields considered.

Other Materials

Some useful calculations have been made of the absorbed dose after attenuation of neutron beams by slabs of various materials, with the neutron beams incident at various angles to the slab normals (Allen and

TABLE 7—*Material compositions*

Material	Density (g/cm^3)	Elements Contained	Atoms/cm^3 ($\times 10^{-21}$)
Borated polyethylene (8% B_4C by weight)[a]	0.97	H	76.80
		C	39.20
		B^{10}	0.658
		B^{11}	2.67
Water	1.00	H	66.90
		O	33.45
Concrete	2.26	H	13.75
		O	45.87
		Al	1.743
		Si	20.15
NTS[b] soil (dry)	1.15	H	8.553
		O	22.68
		Al	2.014
		Si	9.533
NTS[b] soil (100% sat.)	1.25	H	16.87
		O	27.00
		Al	1.976
		Si	8.963

[a] Several calculations were made for pure polyethylene slabs of density 0.925 g/cm^3 up to 6 inches thick. Results differ negligibly from corresponding results for 8% borated polyethylene.

[b] Nevada test site.

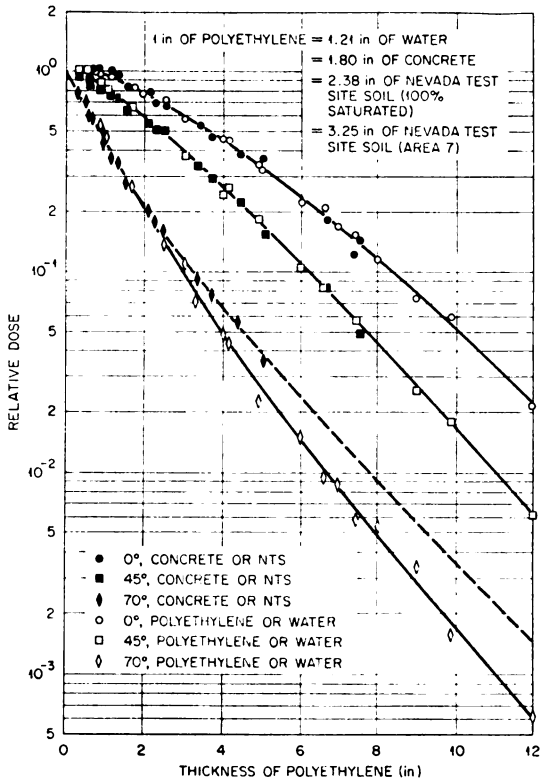

Fig. 61. Attenuation of dose from 5 MeV incident neutrons penetrating various materials at various angles.

Futterer, 1963). The materials reported on are those described in Table 7. The results of the calculations are shown in Figures 61 to 65 which apply, respectively, to incident neutron energies of 5, 3, 2, 1 and 0.5 MeV. In each of these figures the following key applies.

●, 0° incidence, concrete and NTS[14]
■, 45° incidence, concrete and NTS
◆, 70° incidence, concrete and NTS
○, 0° incidence, polyethylene or water
□, 45° incidence, polyethylene or water
◇, 70° incidence, polyethylene or water

Also, in each figure, the abscissa is scaled in units of polyethylene thickness, but a conversion key is provided to re-interpret the scale in terms of thicknesses of the other materials.

No evaluation of error has been provided with these data. However, those portions which appeared unreliable have been omitted.

[14] NTS is Nevada Test Site soil.

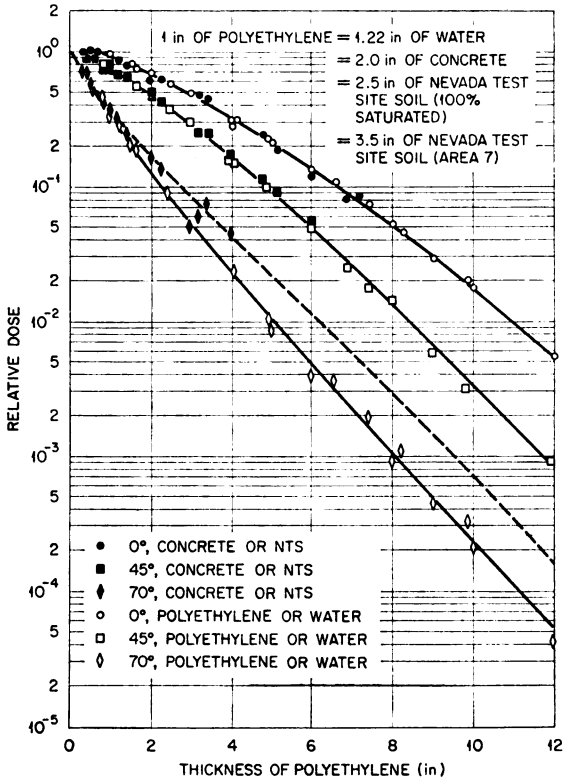

Fig. 62. Attenuation of dose from 3 MeV incident neutrons penetrating various materials at various angles.

To use these curves, one must know the absorbed dose at one surface of a slab of one of these materials due to neutrons incident in a broad beam at an angle (or angle band) and energy (or energy band) close to an angle and energy for which attenuation data are here reported. The attenuation factor appropriate to the material, thickness, energy, and angle is read from the charts, and the incident dose multiplied by that factor should approximate the dose on the other side of the slab.

Removal Cross Sections

Removal cross-section theory permits estimation of neutron shielding in a simple "one velocity" model. The requirements that must be met for removal cross-section theory to apply are:

1. The shield must be sufficiently thick and the neutrons so dis-

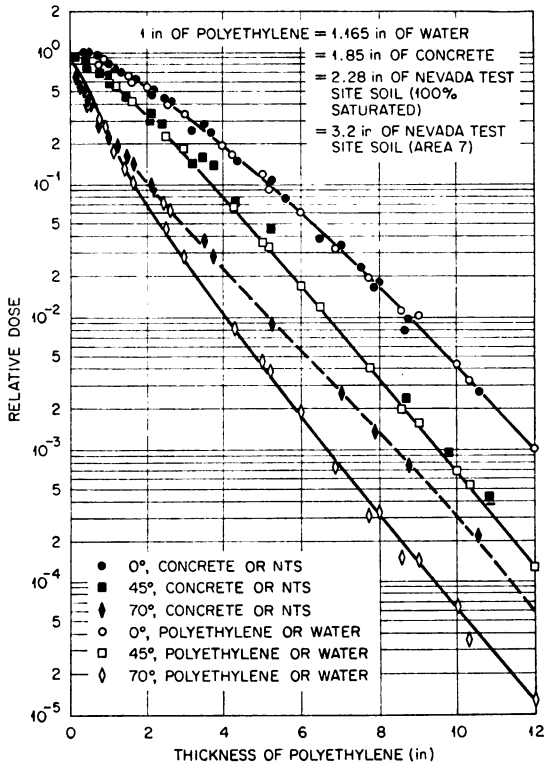

Fig. 63. Attenuation of dose from 2 MeV incident neutrons penetrating various materials at various angles.

tributed in source energy that only a narrow band of the most penetrating source neutrons gives any appreciable ultimate contribution to the dose outside the shield.

2. There must be sufficient hydrogen in the shield—intimately mixed or in the final shield region—to assure a very short characteristic transport length from about 1 MeV to absorption at or near thermal energy.
3. The source energy distribution and the shield material (nonhydrogenous) properties must be such as to assure a short transport distance for slowing down from the most penetrating energies to 1 MeV.

Requirements 2 and 3 assure that spatial equilibrium of all other components with the most penetrating component will be rapidly approximated. Requirement 1 assures that the purely material attenuation of the dose will be exponential. Measurements of removal cross sections

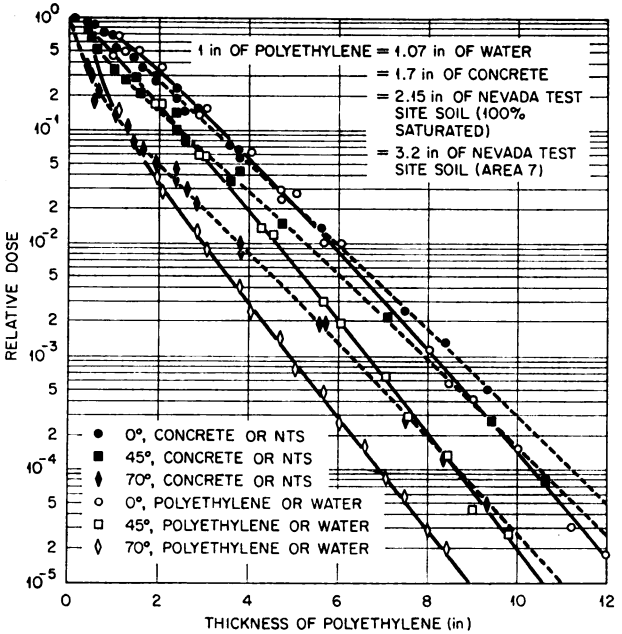

Fig. 64. Attenuation of dose from 1 MeV incident neutrons penetrating various materials at various angles.

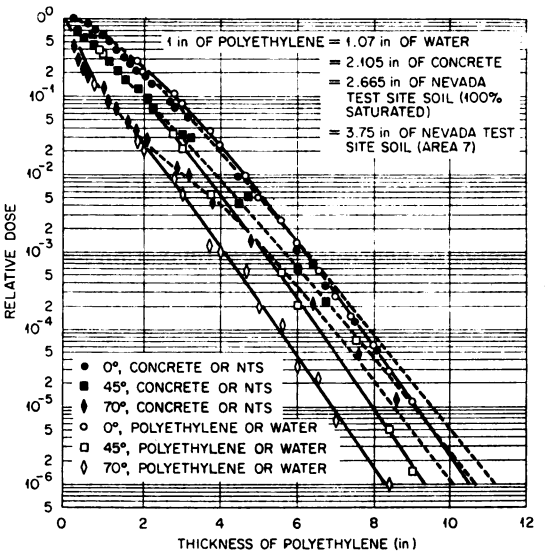

Fig. 65. Attenuation of dose from 0.5 MeV incident neutrons penetrating various materials at various angles.

have been restricted to situations with fission sources and, generally, for shields so thick that the 6–8 MeV source component dominated the dose. In such application, one may write for the dose rate outside the shield

$$\dot{D} = PNfCG\, e^{-(\Sigma t)_{TOTAL}} \text{ mrad/h} \qquad \text{(E5)}$$

P = power level in kilowatts
N = fission neutrons produced per second per kilowatt
$= 7.5 \times 10^{12}$
f = fraction of source neutrons effective in producing dose (requirement 1) $\approx .01$
C = (conversion factor) neutrons cm^{-2} s^{-1} to mrad/h (absorbed dose rate in tissue)
$\approx 1.55 \times 10^{-2}$
G = geometric form factor taking into account distribution of source points and distance from source point to point where dose is determined
$(\Sigma t)_{TOTAL}$ = total number of removal relaxation lengths separating effective source point from dose point.

With the values given above for N, f, and C, equation (E5) becomes

$$\dot{D} = 1.16 \times 10^{10}\, PG\, e^{-(\Sigma t)_{TOTAL}}$$

Some of the geometric factors, G, are as follows

Geometry	Factor
Point source	$\frac{1}{4\pi r^2}$
Parallel Beam	1

(r = separation of source point and dose point in centimeters)

A variety of form factors applying to complicated geometries can be found in the literature. (Price et al., 1957; Rockwell, 1956.)

The total number of neutron removal relaxation lengths along the ray from source point to dose point is the sum of the removal relaxation lengths along the path of the ray in each region traversed by the ray. That is

$$(\Sigma t)_{TOTAL} = \Sigma_{r1} t_1 + \Sigma_{r2} t_2 + \cdots \qquad \text{(E6)}$$

Σ_{ri} = macroscopic removal cross section of region i
t_i = distance traversed by source-dose point ray in region i.

The macroscopic removal cross-section is given by

$$\Sigma_r = \frac{0.602\, \sigma_r \rho}{A}\ (cm^{-1}), \qquad \text{(E7)}$$

where

σ_r = microscopic removal cross section (barns),

ρ = density (g/cm³),

A = atomic weight.

It will be noted that Σ_r/ρ is a quantity dependent only on microscopic nuclear properties. This is a smoothly varying function of the atomic weight, which is given in Figure 66. From this curve the removal cross section can be obtained for any element.

The macroscopic removal cross section for a material of several elements is obtained by simple summation over its constituents:

$$\Sigma_r, \text{compound} = \left(\frac{\Sigma_r}{\rho}\right)_1 \rho_1 + \left(\frac{\Sigma_r}{\rho}\right)_2 \rho_2 + \cdots, \qquad \text{(E8)}$$

where

$\left(\frac{\Sigma_r}{\rho}\right)_1$ = value from Figure 66 for element 1 of compound (cm²/g), etc.

ρ_1 = density of element 1 (g/cm³), etc.

Example: Find Σ_r for $CaCO_3$; density = 2.711, molecular weight = 100.09.

Element	A_i	ρ_i	$(\Sigma_r/\rho)_i$	$(\Sigma_r/\rho)_i\rho_i$
		g/cm³	cm²/g	cm⁻¹
Ca	40.08	$\frac{40.08}{100.09} \times 2.711 = 1.087$	0.0242	0.0263
C	12.01	$\frac{12.01}{100.09} \times 2.711 = 0.325$	0.0510	0.0166
O	16	$\frac{3 \times 16}{100.09} \times 2.711 = 1.299$	0.0410	0.0533
Total				0.0962

$$\Sigma_r CaCO_3 = 0.0962 \text{ cm}^{-1}$$

The removal cross sections for some common shielding and reactor materials are given in Table 8.

There are, of course, many inherent oversimplifications in Equation (E5) which severely limit its accuracy so that it is usually sufficient only for a rough estimation of absolute dose values. It is a much more effective algorithm for comparing an unknown situation with a known. If subscript 1 represents the unknown and 2 the known,

$$\frac{\dot{D}_1}{\dot{D}_2} = \frac{P_1 N_1 f_1 C_1 G_1 e^{-(\Sigma t)_{\text{TOTAL}_1}}}{P_2 N_2 f_2 C_2 G_2 e^{-(\Sigma t)_{\text{TOTAL}_2}}}$$

One need then only consider such quantities as

$$\frac{N_1 f_1 C_1}{N_2 f_2 C_2}, \frac{G_1}{G_2}, \text{ and } e^{(\Sigma t)_{\text{TOTAL}_2} - (\Sigma t)_{\text{TOTAL}_1}}$$

ρ_1/ρ_2 should be known.

If the situations are indeed reasonably comparable,

$$\frac{N_1 f_1 C_1}{N_2 f_2 C_2} \approx 1,$$

$$\frac{G_1}{G_2} \approx 1 \text{ or a simple function}$$

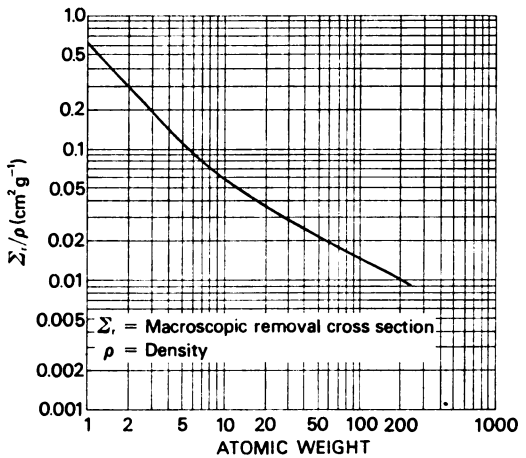

Fig. 66. Fission neutron attenuation coefficients.

TABLE 8—*Macroscopic neutron removal cross sections* (Σ_r) *and attenuation lengths of some shielding and reactor materials*

Material	Σ_r, Removal Cross Section	Attenuation Length
	cm^{-1}	cm
Water	0.103	9.7
Iron	0.1576	6.34
Ordinary concrete[a]	0.089	11.3
Graphite (ρ = 1.54 g cm^{-3})	0.0785	12.7

[a] The removal cross section of barytes concrete will not differ significantly from the removal cross section of ordinary concrete with similar water content and sand and mortar mix. The concrete removal cross section is, however, very sensitive to water content. The value quoted above is for 6 percent water by weight.

of source-dose point separations, and

$$(\Sigma t)_{\text{TOTAL}_2} - (\Sigma t)_{\text{TOTAL}_1}$$

should be fairly small and probably less uncertain than either $(\Sigma t)_{\text{TOTAL}}$ taken separately.

There have been efforts to generalize the removal cross section concept to a multivelocity theory, that is, to remove requirement 1 (Avery et al., 1960). These efforts have met with success where adequate data were provided. For general application, it would be necessary to provide a body of measured empirical data of scope comparable to that available for the one velocity model.

The one velocity removal cross section concept could be extended to energies somewhat higher than the fission spectrum range, in particular to the important 14 MeV range, provided the three requirements are met and provided that removal cross sections applying to that range are measured or otherwise made available. If we limit our attention to the 14 MeV range, it is reasonable to expect the three requirements to be met by moderately thick shields containing at least a final hydrogenous region of moderate thickness. Depending on the material involved, the appropriate removal cross section can be less or greater than the corresponding cross section for fission neutrons. (Compare Figures 50, 53 and 54. They clearly show that for concrete the removal cross section at 14 MeV is greater than it is for fission neutrons.)

Other Aids

There are many other aids to neutron shielding calculations too elaborate to summarize here. An extensive survey of neutron reflection has been carried out by Maerker and Muckenthaler (1965). There are any number of special and general purpose high speed computing machine programs available at various laboratories, or from the Radiation Shielding Information Center at the Oak Ridge National Laboratory.

APPENDIX F

Accident Dosimetry[15]

Quick-sort Methods

The first step in the sequence of procedures leading to a dose determination is the positive identification of exposed persons by some type of "quick-sort" procedure.

An indium foil (0.5 to 1 g) worn on the person is a highly satisfactory indicator of neutron irradiation, since it is sufficiently sensitive to permit activation measurements with ion chamber as well as GM and scintillation instruments. Frontal exposure data (Wachter and Emerson, 1956) using unmoderated fission neutrons leads to a sensitivity figure, based on contact gamma readings with an ionization chamber survey meter, of 5 mR/h per 1000 ergs/g neutron kerma for tissue in free space. This figure pertains only to the activity induced in an indium foil (0.9 g) by direct source neutrons reflected from a phantom. The relationship of foil activity to total dose is so strongly dependent on such factors as orientation, neutron spectrum, and gamma/neutron ratio that it cannot, in general, be made the basis for even an approximate dose estimate. In the absence of definitive information regarding these factors, it is advisable to consider a person significantly exposed if he is wearing an indium foil with "detectable" induced activity. Because of the short (54 min) half-life of indium, the check should be made as soon as possible after exposure.

An alternative method of checking for neutron irradiation employs a gamma survey instrument for a midsection surface measurement of radioactivity induced in the body. Although the much lower sensitivity precludes the use of ion chamber instruments, this method has the advantage of less dependence on orientation than the indium foil procedure. An approximate criterion for whole body exposure to unmoderated fission neutrons may be inferred from measurements on phantoms (Peabody, 1963 and Ballinger et al., 1962) and accident victims (Brittan et al., 1953; Langham, 1961; Lushbaugh, 1961 and UCNC, 1958). Thus

[15] The NCRP is greatly indebted to L. L. Anderson for this contribution.

for a neutron irradiation characterized by a tissue kerma in free air of 1000 ergs/g, the reading associated with body ^{24}Na activity (15 h half-life) will be 1–3 times that produced by a typical level (20 μR/h) of background gamma radiation. A reading taken immediately after a short exposure would be higher by a factor of 2, due to the ^{38}Cl activity (37 min half-life) that is present. A higher reading per unit kerma will also be observed if the spectrum is degraded or if the counting geometry is improved by having the exposed person bend over the survey probe (Wilson, 1962). It is clearly desirable to apply the quick-sort procedure in a low-background area to persons known to be free from contamination.

Analysis of Biological Materials

Both blood and whole body measurements of ^{24}Na activity are important in the more accurate assessment of absorbed dose. Such measurements provide a dose normalization factor which is relatively independent of movement during exposure. Phantom studies (Sanders and Auxier, 1962) have indicated that the specific activity ($\mu Ci^{24}Na/g^{23}Na$) in blood is closely equal to the average specific activity throughout the body. Because of variability in estimates of total body sodium (ICRP, 1960; Smith et al., 1965; and Brittan et al., 1953), the measurement of specific activity in blood may be more reliable than the corresponding whole-body measurement.

In experiments with burros (Sanders and Auxier, 1962), sodium specific activity has been shown to depend somewhat on body weight but to be an insensitive function of blood sampling time up to 24 hours after exposure, provided the usual radioactive decay correction is made. For later blood sampling times, a correction, based on a biological half-life of eleven days (Dennis, 1964) may be applied. Sodium activation has been observed (Sanders and Auxier, 1962; Anderson et al., 1964) to vary with orientation by as much as 36 percent, but presumably the average absorbed dose to the body varies in the same direction.

Dosimetric interpretation of blood or whole body ^{24}Na measurements requires knowledge of the neutron spectrum. The kerma (for tissue in air) per unit sodium activation is given by the ratio

$$\frac{\int_0^\infty K(E)\phi(E)\mathrm{d}E}{\int_0^\infty \xi(E)\phi(E)\mathrm{d}E}$$

where $\phi(E)$ is the neutron fluence per unit energy at energy E, $K(E)$ is the kerma per unit fluence, and $\xi(E)$ is the specific activity of ^{24}Na per unit fluence in the person exposed. Similarly, an absorbed dose per unit activation may be represented by substituting for $K(E)$ a function $D(E)$ defined, for example, as the surface dose or the maximum depth dose, and developed from data presented elsewhere in this report.

The function $\xi(E)$, for phantoms, has been calculated (Hurst et al., 1959; Auxier et al., 1961; ORNL, 1965) at Oak Ridge National Laboratory and measured (Smith, 1962) at Harwell, with fair agreement. It is proportional to the total neutron capture probability, except above 3 MeV where threshold reactions occur. The recent calculations (ORNL, 1965) suggest that resonance capture is not negligible, contrary to earlier opinion. Average values of $\xi(E)$, in units of 10^{-12} μCi $^{24}Na/g^{23}Na$ per neutron/cm^2, have been given (Smith et al., 1965) as 4.1 for fission neutrons, 2.9 for intermediate energy neutrons, and 1.6 for thermal neutrons.

In certain cases, it may suffice to infer knowledge of the neutron spectrum by comparing the accident situation with previous accidents or experimental configurations which have been extensively studied. Then a direct estimate of neutron kerma is possible, using values of blood sodium specific activity derived from published data. Typical values, in units of $\mu Ci^{24}Na/g^{23}Na$ per erg/g, are 2.0–2.5 $\times$ 10^{-5} for the Godiva II reactor (Sanders and Auxier, 1962; Ritchie and Eldridge, 1961), 3.2 $\times$ 10^{-5} for the Oak Ridge Y-12 accident (Hurst et al., 1959; Ritchie and Eldridge, 1961), and 1.2 $\times$ 10^{-4} for the Yugoslav accident (Hurst et al., 1961). When the neutron spectrum is highly degraded, as in the Yugoslav accident, the absorbed dose obtained by a simple conversion (ergs/g to rads) from neutron kerma may be much lower than the actual surface dose from neutrons, since the latter includes a significant contribution from neutron capture gamma rays (Hurst et al., 1961).

Sodium activity in the body may serve also as the basis for an estimate of gamma-ray kerma, provided the ratio of gamma-ray kerma to neutron kerma has been determined separately. This consideration is especially important if no gamma-ray dosimeter was worn, or if orientation uncertainties make interpretation difficult. Values of the γ/n kerma ratio for the three critical assemblies mentioned above are, in the same order, 0.22, 2.8, and 3.5–4.2 (Hurst et al., 1959; Ritchie and Eldridge, 1961; Hurst et al., 1961).

Measurements of ^{32}P activity in body hair have been employed to good advantage in the evaluation of accidental neutron exposures (Harris, 1961; Petersen, 1963; HLC, 1963). Such measurements, using

samples from different parts of the body, are helpful in establishing orientation and in evaluating the spatial distribution of K_S, the kerma associated with neutrons of energy greater than 2.9 MeV, the threshold of the $^{32}S(n,p)^{32}P$ reaction. Interference from the $^{31}P(n,\gamma)^{32}P$ reaction is negligible if the ratio of thermal to above threshold fluence is less than 10, i.e., the reaction cross sections are similar and the concentration of phosphorus in hair is less than 1 percent that of sulfur. For unfavorable fluence ratios, a correction may be applied. The sensitivity of the method extends to K_S values as low as 500 erg/g for 1 g samples (Petersen et al., 1961). K_S, in erg/g, is roughly 50 times the ^{32}P activity in dpm per gram of hair (Petersen, 1965).

Other biological materials have been used or suggested for measurements in connection with accidental neutron exposure. Urinary ^{32}P and ^{24}Na activities have been measured but have proved of little value in dosimetry (Langham, 1961; Parker and Newton, 1965). Analysis for ^{32}Si and ^{32}P activities in bone biopsy material has been proposed as a means of measuring separately the fast and thermal components of neutron fluence, making use of the (n,p) and (n,γ) reactions in bone phosphorus (Chanteur et al., 1965).

Analysis of Other Materials Exposed During Accident

A rapid estimate of the neutron spectrum, following an accidental exposure, generally requires data from a previously prepared system of activation materials. Activation detectors have the advantages of inherent discrimination against gamma rays, linear response even for very high flux densities, and long-term, service-free monitoring capability. A threshold detector unit developed at Oak Ridge for spectral measurements near critical assemblies has been described in other NCRP reports (NCRP, 1960a; NCRP, 1961). It employs ^{239}Pu, ^{237}Np, and ^{238}U foils in a ^{10}B sphere, together with a sulfur pellet, to permit determining fluences above successive threshold energies of 1 keV, 0.75 MeV, 1.5 MeV, and 2.9 MeV. Gold foils provide for a cadmium difference measurement of thermal neutrons. The method of interpreting data from this system has since been reformulated so that it takes account also of neutrons at energies less than 1 keV (Bailey, 1964; Anderson et al., 1965).

The expense of ^{10}B spheres and possible difficulty in obtaining fission foils are avoided, with some sacrifice of energy resolution, in foil activation systems using only commonly available materials (Smith et al., 1965; Reinig et al., 1965; Bramson, 1962; Bricka and Cercy, 1965).

Threshold reactions that have been utilized include $^{115}In(n,n')^{115m}In$ for $E_n > 1$ MeV, $^{32}S(n,p)^{32}P$ for $E_n > 2.9$ MeV, and $^{24}Mg(n,p)^{24}Na$ for $E_n > 6$ MeV. Cadmium-covered indium, gold and copper foils have resonances suitable for use in the intermediate energy region (0.5 eV to 10 keV); their activation data may be combined to yield an index of the resonance neutron spectrum (Anderson, 1964; Candes and Lavie, 1965). The persistent gap in activation foil response from 10 keV and 1 MeV may be partially filled by using a moderated foil detector (Candes and Lavie, 1965).

Schemes for deploying activation foil units have generally followed the plan set forth by Hurst and Ritchie in 1959 (Hurst and Ritchie, 1959), i.e., each potential accident area is monitored by a primary station with full neutron spectrum-measuring capability, and several secondary stations which serve to define the pattern of radiation and to provide data for spectrum corrections. One or more gamma-ray dosimeters are customarily included in activation foil packages. It is thus possible to compute the γ/n kerma ratio at the location of each unit.

It is usual practice to include activation foils in monitoring devices worn by personnel, as well. In interpreting data from such foils, it is necessary to take account of the moderating and attenuating effects of the body. For thermal and resonance energies, neutrons reflected from the body may be distinguished from those incident upon it by making use of beta counting data from two gold foils separated by a cadmium absorber (Adams and Dennis, 1962). The orientation of the exposed person must be known, however, to permit full use of activation data from personnel dosimeters. An experimental evaluation of orientation is possible if the person exposed was wearing a belt containing four or more activation units at fixed intervals (Smith et al., 1965; Hurst and Ritchie, 1959; Auxier et al., 1961). The need for orientation information may be avoided in part by installing, as monitors, body phantoms each filled with a sodium-bearing material and surrounded by several neutron foil units (Smith et al., 1965). The kerma (or absorbed dose) per unit sodium activation can then be determined for the phantom and applied directly in computing the kerma (or absorbed dose) to a person, from his blood sodium activity.

Installed monitoring devices that measure neutron kerma directly are of limited utility in the dose determination based on body sodium activity, which requires spectral information for normalization. If the person's location at the time of exposure is known, however, the presence of such monitors, in a number sufficient to define the field, will permit an independent estimate of kerma.

Neutron kermametric devices with the gamma discrimination capa-

bility, rate independence, and range required for installed monitors have not, until recently, been generally available. The development of silicon diode dosimeters which meet these criteria has been in progress for several years (Endres et al., 1962; Thursten et al., 1967). In the energy region 0.45 to 4.0 MeV a response proportional to kerma for tissue within ±10 percent may be inferred from reported data. Thermoluminescent neutron detectors also have response-vs-energy characteristics governed by the reaction cross sections of their constituent elements (Kastner et al., 1966), and caution must be exercised in using them to estimate kerma for tissue. A promising approach for accident monitoring employs a thermoluminescent material of low intrinsic neutron sensitivity (e.g., ^{7}LiF) in intimate contact with a hydrogenous tissue-like substance (Endres, 1965). The sample is paired with one containing no hydrogen in order to allow a subtraction of gamma response. Further energy dependence studies of this system are needed, with emphasis on the effect of particle size in the mixture.

Other neutron devices that have been proposed for direct kermametric or dosimetric monitoring include moderated thermal neutron detectors (Bramson, 1962; Distenfeld et al., 1965), with moderator thickness selected to approximate the desired response function, and fission fragment damage dosimeters, in which observable radiation damage occurs in a sheet of mica, plastic or glass in contact with fissionable material (Fleischer et al., 1965; Unruh et al., 1967; Becker, 1966). The latter technique can be applied also to threshold detector systems for spectrum measurement (Kerr and Strickler, 1966). Nuclear track emulsions are not generally useful for the radiation levels considered here, since darkening caused by gamma rays will, in many cases, prevent evaluation of the proton recoil tracks.

Intercomparison data have shown agreement within about ±25 percent among several accident dosimetry systems employing techniques mentioned above (Auxier, 1967).

A variety of other items present at the time of an accident may yield information helpful for the subsequent evaluation. Recorder charts from area gamma-ray monitors may assist in establishing the time of a radiation burst and may, in some instances, provide normalizing data as well. Tools, apparatus, and other objects containing chemical elements such as Au, Cu, Ag, Al, In, Cd, Ni, Mn, Zn, Sn, Sb, Hg, Ge, Si, Na, and Ba may serve as neutron monitors if their induced radioactivity can be measured at known and preferably short times after the exposure. Personal effects containing activation materials should be collected and analyzed, although their usefulness may depend on the accuracy with which location and orientation can be determined. Most induced radio-

activities will result predominantly from thermal neutrons, so that post-accident calibrations of exposed objects may be carried out by exposure to known thermal fluences. In some cases, estimates of fast neutron fluence may be possible from such threshold reactions as $^{58}Ni(n,p)^{58}Co$ (in coins) and $^{31}P(n,p)^{31}Si$ (in non-safety matches). Facilities for gamma-ray spectrometry should be utilized to the greatest extent possible, since separate identification of multiple activities may prove essential to the analysis. In the case of a criticality accident, the total number of fissions is of interest and can be determined by radiochemical analysis of the reaction material for fission products of known yield, e.g., ^{99}Mo (Lyon et al., 1962).

Dosimetry of Reconstructed Accident

In the event that information on the neutron spectrum and γ/n ratio is not available, either from installed monitors or previous measurements, an experiment to duplicate the conditions of the accident may be indicated. In carrying out such an experiment, it is important to reproduce as closely as possible both the composition of radiations emitted by the source and the scattering and shielding environment of the location at which the person was exposed. Normalization of neutron intensities may be checked using radioactivities induced in objects that were exposed during the accident and later evaluated, although body sodium activation should remain the primary basis for dose computation.

Since low intensities are usually possible in the simulation of an accident, the choice of measuring instruments is not limited by the range and rate dependence criteria that might apply to the accident itself. Thus, neutron spectral measurements may be made with such detectors as nuclear track emulsions (Sanna and O'Brien, 1963; Lehman et al., 1964), proton-recoil scintillators, ^{3}He proportional counters (Sayres and Coppola, 1964), and multiple polyethylene spheres (Bramblett et al., 1960), as well as with activation foil systems. Direct kerma measurements of neutrons are possible using tissue equivalent ionization chambers, film dosimeters, and Bragg-Gray type proportional counters. Instruments with low neutron sensitivity, for the measurement of gamma-ray kerma, include carbon-CO_2 ionization chambers and specially-shielded G-M counters (Wagner and Hurst, 1961). Most of the aforementioned instruments and their associated measurement techniques are described elsewhere in this report or in previously published NCRP reports.

A complete mock-up experiment should provide sodium activation data for a man-shaped phantom, or for an experimental animal, at the presumed location of the accidental exposure. These data, together with spectral and direct kerma measurements at the same location, will furnish experimental values of kerma (or absorbed dose) per unit sodium activation, without requiring an assumption concerning the shape of the activation function, $\xi(E)$.

The measured neutron spectrum itself will permit separate evaluation of the ratio of neutron kerma (or absorbed dose) to body sodium activation, if published values are utilized for the functions $\xi(E)$ and $K(E)$, or $D(E)$. A γ/n ratio may require correction, on a theoretical basis, to take account of (1) the relevant difference between constant power and burst exposures and (2) the itinerary followed by the person in the original delayed gamma-ray field.

Finally, a completely theoretical evaluation of the kerma (or absorbed dose) per unit sodium activation may be undertaken by first calculating the neutron spectrum and the γ/n ratio from the physical parameters of the source. Excellent guidance in both the theoretical and experimental aspects of absorbed dose assessment via reconstruction is available from published studies of previous accidents (Brittan et al., 1953; Hurst et al., 1959; Hurst et al., 1961; Hasterlik and Marinelli, 1956).

References

Specific References[16]

ADAMS, N. AND DENNIS, J. A. (1962). "A new method of using gold foils for the investigation of the leakage spectra from critical assemblies," *Neutron Dosimetry*, Vol. I, p. 173 (Int. Atomic Energy Agency, Vienna, 1962)

ALLEN, F. J. AND FUTTERER, A. T. (1963). "Neutron Transmission Data," Nucleonics, **21**, No. 8, 120 (1963)

AMALDI, E. (1959). "The production and slowing down of neutrons," *Encyclopedia of Physics*, 38/2 (Springer-Verlag, New York, 1959)

ANDERSON, I. O. AND BRAUM, J. (1964). "A neutron rem counter," Nucleonik **6**, 237

ANDERSON, J., OSBORN, S. B., TOMLINSON, R. W. S., NEWTON, D., RUNDO, J., SALMON, L. AND SMITH, J. W. (1964). "Neutron-activation analysis in man in vivo," Lancet **2**, 1201

ANDERSON, L. L. (1964). "Cross section of copper-63 for nuclear accident dosimetry," Health Physics **10**, 315 (1964)

ANDERSON, L. L., DUFFY, T. L., SEDLET, J. AND O'NEIL, D. P. (1965). "Nuclear accident dosimetry at Argonne National Laboratory". *Personnel Dosimetry for Radiation Accidents*, p. 645 (Int. Atomic Energy Agency, Vienna, 1965)

ATTIX, F. H. AND ROESCH, W. C. eds. (1966). *Radiation Dosimetry*, Second edition, *Vol. II, Instrumentation* (Academic Press, New York, 1966)

ATTIX, F. H. AND ROESCH, W. C. eds. (1968). *Radiation Dosimetry*, Second edition, *Vol. I, Fundamentals* (Academic Press, New York, 1968)

ATTIX, F. H. AND TOCHILIN, E. eds. (1969). *Radiation Dosimetry*. Second edition. *Vol. III, Sources, Fields, Measurements, and Applications* (Academic Press, New York, 1969)

AUXIER, J. A. (1967). "Multilaboratory intercomparisons of neutron dosimetry systems", *Neutron Monitoring*, p. 625 (Int. Atomic Energy Agency, Vienna, 1967)

AUXIER, J. A., SANDERS, F. W. AND HENSLEY, P. N. (1961). "A device for determining the orientation of persons exposed to neutron and/or γ radiation", Health Physics **5**, 226 (1961)

AUXIER, J. A., SANDERS, F. W. AND SNYDER, W. S. (1961). "^{24}Na activation in the dosimetry of nuclear accidents", *Radioactivity in Man*, G. G. Meneely, ed., p. 201 (Charles C Thomas, Springfield, Ill., 1961)

AUXIER, J. A., SNYDER, W. S. AND JONES, T. D. (1968). "Neutron interactions

[16] Information on the availability of NCRP Reports listed is given on pages 146–148.

and penetration in tissue", *Radiation Dosimetry*, 2nd edition. Vol. I, F. H. Attix and W. C. Roesch, eds. p. 275 (Academic Press, New York, 1968)

AVERY, A. F., BENDALL, D. E., BUTLER, J. AND SPINNEY, K. T. (1960). *Methods of calculation for use in the design of shields for power reactors*, Report AERE-R-3216 (H. M. Stationery Office, London, 1960)

BAILEY, J. C. (1964). *Criticality-accident dosimetry studies*, AEC Report No. K-1618, U.S. Atomic Energy Commission, Washington, 1964)

BALLINGER, E. R., HARRIS, P. S., CARR, L., HIEBERT, R. AND LARKINS, J. (1962). "Body ^{24}Na activity as a measure of neutron dose", Nucleonics **20**, No. 10, 76 (1962)

BARTHOLOMEW, G. A., GROSHEV, L. V., DOVEIKA, A., EASTWOOD, K. M., MONARO, S., DEMIDOV, A. M., PELEKHOV, V. I. AND SOKOLOSKII, L. L. (1967). "Compendium of Thermal-Neutron-Capture γ-Ray Measurements. Part I. $Z \leqslant 46$", *Nuclear Data*, A3, p. 367 (Academic Press, New York, 1967)

BECKER, K. (1966). "Nuclear track registration in dosimeter glasses for neutron dosimetry in mixed radiation fields", Health Physics **12**, 769 (1966)

BERTINI, H. W. (1963). "Low-energy intranuclear-cascade calculations", Phys. Rev. **131**, No. 4, 1801 (1963)

BERTINI, H. W. (1965). *Results from low-energy intranuclear-cascade calculation*, ORNL-TM-1225 (Oak Ridge National Laboratory, Oak Ridge, Tennessee, 1965)

BETHE, H. AND ASHKIN, J. (1964). *Experimental Nuclear Physics*. Vol. I, Part II, E. Segre, Ed., p. 166 (John Wiley and Sons, New York, 1964)

BNL (1958). HUGHES, D. J. AND SCHWARTZ, R. B., *Neutron cross sections*, BNL-325, 2nd edition (Brookhaven National Laboratory, Upton, Long Island, New York, 1958)

BNL (1964), STEHN, J. R., GOLDBERG, M. D., MAGURNO, B. A. AND WEINER-CHASMAN, R., *Neutron cross sections*, Vol. I. BNL-325, Supplement No. 2, (Brookhaven National Laboratory, Upton, Long Island, New York, 1964)

BRAMBLETT, R. L., EWING, R. I. AND BONNER, T. W. (1960). "A new type of neutron spectrometer", Nucl. Inst. and Methods **9**, 1 (1960)

BRAMSON, P. E. (1962). *Hanford criticality dosimeter*, HW-71710 Rev. (General Electric Co., Hanford Atomic Products Operation, Richland, Washington, 1962)

BRICKA, M. AND CERCY, J. (1965). "Le spectrometre neutrons à activation (SNAC) et sa réponse sur modérateur", *Personnel Dosimetry for Radiation Accidents*, p. 381 (Int. Atomic Energy Agency, Vienna, 1965)

BRITTAN, R. O., MARINELLI, L. D. AND THALGOTT, F. W. (1953). *Technical review of ZPR-I accidental transient-power excursion and exposures*, ANL-4971 Rev. (Argonne National Laboratory, Argonne, Ill., 1953)

BRUSTAD, T. (1961). "Effects of fast charged particles," Rad. Res. **15**, 139 (1961)

CANDES, P. AND LAVIE, J. M. (1965). "La dosimétrie des radioexpositions accidentelles externes au commissariat à l'énergie atomique," *Personnel*

Dosimetry for Radiation Accidents, p. 607 (Int. Atomic Energy Agency, Vienna, 1965)

CHANTEUR, J., ARNAUD, Y. AND PELLERIN, P. (1965). Dosimétrie biologique après exposition à un flux élevé de neutrons thermiques et rapides," *Personnel Dosimetry for Radiation Accidents*, p. 235 (Int. Atomic Energy Agency, Vienna, 1965)

CLARK, F. H., BETZ, N. A. AND BROWN, J. (1966). *Monte Carlo calculations of the penetration of normally incident neutron beams through concrete*, ORNL-3926 (Oak Ridge National Laboratory, Oak Ridge, Tennessee, 1966)

DENNIS, J. A. AND LOOSEMORE, W. R. (1960). *A fast neutron counter for dosimetry*, Report AERE-R-3302 (H. M. Stationery Office, London, 1960)

DENNIS, J. A. (1964). *Dosimetry in criticality accidents*, Report AERE-R-4365 (H. M. Stationery Office, London, 1964)

DISTENFELD, C. H., BISHOP, W. H. JR. AND COLVETT, D. (1965). *A thermoluminescent neutron dosimetry system*, BNL-9605 (Brookhaven National Laboratory, Upton, Long Island, New York, 1965)

DRESNER, L. (1962). *EVAP—A Fortran program for calculating the evaporation of various particles from excited compound nuclei*, ORNL-TM-196 (Oak Ridge National Laboratory, Oak Ridge, Tennessee, 1962)

ENDRES, G. W. R., BRAMSON, P. E. AND MEDCALF, A. W. (1962). *Silicon semiconductor neutron dosimeter* (General Electric Company, Hanford Atomic Products Operation, Richland, Washington, 1962)

ENDRES, G. W. R. (1965). *Thermoluminescent dosimetry studies at Hanford*, BNWL-SA-23 (Battelle Northwest, Richland, Washington. Pacific Northwest Laboratory, 1965)

ENGLE, W. W., JR. (1967). *A users manual for ANISN: A one dimensional discrete ordinates transport code with anisotropic scattering.* (K-1693) (Oak Ridge Gaseous Diffusion Plant, Oak Ridge, Tennessee, 1967)

ENEA (1963). *Personnel dosimetry techniques for external radiation, their application in nuclear installations, Proc. of a Symposium, Madrid, 1963* (European Nuclear Energy Agency, Organization for Economic Cooperation and Development, Brussels, 1963)

FLEISCHER, R. L., PRICE, P. B. AND WALKER, R. M. (1965). "Solid state track detectors; applications to nuclear science and geophysics," Ann. Rev. Nucl. Sci. **15**, 1 (1965)

GROSHEV, L. V., DEMIDOV, A. M., PELEKHOV, V. I., SOKOLOSKII, L. L., BARTHOLOMEW, G. A., DOVEIKA, E., EASTWOOD, K. M. AND MONARO, S. (1968). "Compendium of Thermal-Neutron-Capture γ-Ray Measurements. Part II. $Z = 47$ to $Z = 67$ (ag to Ho)," *Nuclear Data*, A5, p. 1 (Academic Press, New York, 1968)

GROSHEV, L. V., DEMIDOV, A. M., PELEKHOV, V. I., SOKOLOVSKII, L. L., BARTHOLOMEW, G. A., DOVEIKA, A., EASTWOOD, K. M., AND MONARO, S. (1969). "Compendium of Thermal-Neutron-Capture γ-Ray Measurements.

Part III. Z = 68 to Z = 94 (Er to Pu)," *Nuclear Data*, A5, p. 243 (Academic Press, New York, 1969)

HANKINS, D. E. (1963). "Monitoring intermediate energy neutrons," Health Physics **9**, 31 (1963)

HARRIS, P. S. (1961). "Radiation dose estimation in the 1958 Los Alamos criticality accident," Health Physics **5**, 37 (1961)

HASTERLIK, R. J. AND MARINELLI, L. D. (1956). "Physical dosimetry and clinical observations on four human beings involved in an accidental criticality excursion," *Proc. Int. Conf. on Peaceful Uses of Atomic Energy*. Vol. II, p. 25 (United Nations, New York, 1956)

HENDERSON, B. J. (1959). *Conversion of neutron and gamma-ray flux to absorbed dose rate*. Radiation Shielding Course, Oak Ridge School of Reactor Technology, unpublished. (Oak Ridge National Laboratory, Oak Ridge, Tennessee, 1959)

HESS, W. N., PATTERSON, H. W., WALLACE, R. AND CHUPP, E. L. (1959). "Cosmic-ray neutron energy spectrum," Phys. Rev. **116**, 445 (1959)

HINE, G. J. AND BROWNELL, G. L. eds. (1956). *Radiation Dosimetry*. First edition (Academic Press, New York, 1956)

HLC (1963). HANFORD LABORATORIES COMMITTEE; GAMERTSFELDER, C. C., LARSON, H. V., NIELSEN, J. M., ROESCH, W. C. (Chairman), AND WATSON, E. C., "Dosimetry investigation of the Recuplex criticality accident," Health Physics **9**, 757 (1963)

HURST, G. S. AND RITCHIE, R. H., eds. (1959). *Radiation accidents: dosimetric aspects of neutron and gamma-ray exposures*. ORNL-2748, Part A (Oak Ridge National Laboratory, Oak Ridge, Tennessee, 1959)

HURST, G. S., RITCHIE, R. H. AND EMERSON, L. C. (1959). "Accidental radiation excursion at the Oak Ridge Y-12 Plant, III. Determination of radiation doses," Health Physics **2**, 121 (1959)

HURST, G. S., RITCHIE, R. H., SANDERS, F. W., REINHARDT, P. W., AUXIER, J. A., WAGNER, E. B., CALLIHAN, A. D. AND MORGAN, K. Z. (1961). "Dosimetric investigation of the Yugoslav radiation accident," Health Physics **5**, 179 (1961)

IAEA (1961). *Selected topics in radiation dosimetry*. Proc. of a Symposium (International Atomic Energy Agency, Vienna, 1961)

IAEA (1963). *Neutron Dosimetry*, Vols. I and II, Proc. of a Symposium on neutron detection, dosimetry and standardization (International Atomic Energy Agency, Vienna, 1963)

IAEA (1967). *Neutron monitoring*, Proc. of a Symposium (International Atomic Energy Agency, Vienna, 1967)

ICRP (1960). International Commission on Radiological Protecton, "Report of ICRP Committee II on permissible dose for internal radiation (1959), with bibliography for biological, mathematical and physical data," Health Physics **3**, (1960)

ICRP/ICRU (1963). "Report of the RBE Committee," Health Physics **9**, 357 (1963)

ICRU (1964). ICRU Report 10b. *Physical aspects of irradiation*, National

Bureau of Standards Handbook 85 (U. S. Government Printing Office, Washington, 1964)

ICRU (1968). ICRU Report 11, *Radiation quantities and units* (International Commission on Radiation Units and Measurements, Washington, 1968)

Irving, D C., Freestone, R. M., Jr. and Kam, F. B. K. (1965). *05R, A general-purpose Monte Carlo neutron transport code,* ORNL-3622 (Oak Ridge National Laboratory, Oak Ridge, Tennessee, 1965)

Irving, D. C., Alsmiller, R. G., Jr. and Moran, H. S. (1967). "Tissue current-to-dose conversion factors for neutrons with energies from 0.5 to 61 MeV," Nucl. Inst. Methods **51,** 129 (1967) also as ORNL-4302 (Oak Ridge National Laboratory, Oak Ridge, Tennessee, 1967)

Kastner, J., Oltman, B. G. and Tedeschi, P. (1966). "LIF thermoluminescent response to fast neutrons," Health Physics **12,** 1125 (1966)

Kerr, G. D. and Strickler, T. D. (1966). "The application of solid-state nuclear track detectors to the Hurst threshold detector system," Health Physics **12,** 1141 (1966)

Kim, H. J., Milner, W. T. and McGowan, S. K. (1967). "Nuclear cross sections for charged-particle-induced reactions N and O," *Nuclear Data,* A3, p. 123 (Academic Press, New York and London, 1967)

Kinney, W. E. (1964). *The nucleon transport code, NTC.* ORNL-3610 (Oak Ridge National Laboratory, Oak Ridge, Tennessee, 1964)

Langham, W. H. (1961). "Special studies in acute radiation death resulting from an accidental nuclear critical excursion," p. 169, Journal of Occupational Med. **3,** Special Supplement, 147 (1961)

Leake, J. W. (1967). "Portable instruments for the measurement of neutron dose-equivalent rate in steady state and pulsed neutron fields," *Neutron Monitoring,* p. 313 (International Atomic Energy Agency, Vienna, 1967)

Lehman, R. L., Akagi, H. and Fekula, Q. M. (1964). "Neutron dosimetry near a bare pulsed reactor," Health Physics **10,** 517 (1964)

Lindhard, J. and Scharff, M. (1961). "Energy dissipation by ions in the KeV region," Phys. Rev. **124,** 128 (1961)

Lushbaugh, C. C. (1961). "Clinical course of case K," *Acute Radiation Death Resulting from an Accidental Nuclear Excursion,* p. 150, Journal of Occupational Medicine **3,** Special Supplement, 147 (1961)

Lyon, W. S., Reynolds, S. A. and Eldridge, J. S. (1962). "Radiochemistry can help analyze nuclear accidents," Nucleonics **20,** No. 5, 92 (1962)

Maerker, R. E. and Muckenthaler, F. J. (1965). "Calculation and measurement of the fast neutron differential dose albedo for concrete," Nucl. Sci. Eng. **22,** 455 (1965)

Marion, J. F. and Fowler, J. L., eds. (1960). *Fast Neutron Physics.* Vol. I, (Interscience Publishers, Inc., New York, 1960)

Mittleman, P. S. and Liedtke, R. A. (1955). "Gamma rays from thermal-neutron capture," Nucleonics **13,** No. 5, 50 (1955)

Morgan, K. Z. and Turner, J. E., eds. (1967). *Principles of Radiation Protection* (John Wiley and Sons, New York, 1967)

NCRP (1954). NCRP Report No. 17, *Permissible dose from external sources of*

ionising radiation, National Bureau of Standards Handbook 59 (U.S. Government Printing Office, Washington, 1954)

NCRP (1957a). *Maximum permissible radiation exposures. A preliminary statement of the National Committee on Radiation Protection and Measurements*, Insert to accompany National Bureau of Standards Handbook 59 (U.S. Government Printing Office, Washington, 1957)

NCRP (1957b). NCRP Report No. 20, *Protection against neutron radiation up to 30 million electron volts*, National Bureau of Standards Handbook 63 (U.S. Government Printing Office, Washington, 1957)

NCRP (1958). *Maximum permissible radiation exposure to man*, Addendum to National Bureau of Standards Handbook 59 (U.S. Government Printing Office, Washington, 1958)

NCRP (1960a). NCRP Report No. 23, *Measurement of neutron flux and spectra for physical and biological applications*, National Bureau of Standards Handbook 72 (U.S. Government Printing Office, Washington, 1960)

NCRP (1960b). "Statement on maximum permissible dose to the skin of the whole body," Radiology **75**, 122 (1960)

NCRP (1961). NCRP Report No. 25 *Measurement of absorbed dose of neutrons, and of mixtures of neutrons and gamma rays*, National Bureau of Standards Handbook 75 (U.S. Government Printing Office, Washington, 1961)

NCRP (1964). NCRP Report No. 31, *Shielding for high-energy electron accelerator installations*, National Bureau of Standards Handbook 97 (U.S. Government Printing Office, Washington, 1964)

NCRP (1966). NCRP Report No. 32, *Radiation protection in educational institutions* (National Council on Radiation Protection and Measurements, Washington, 1966)

NCRP (1971). NCRP Report No. 39, *Basic radiation protection criteria* (National Council on Radiation Protection and Measurements, Washington, in preparation, 1971)

NORTHCLIFFE, L. C. (1963). "Passage of heavy ions through matter," Ann. Rev. Nucl. Sci. **13**, 67 (1963)

ORNL (1965). "Body sodium activation as a dosimetric tool in nuclear accidents," *Health Physics Division Annual Progress Report for period ending July 31, 1965*, p. 163, ORNL-3849 (Oak Ridge National Laboratory, Oak Ridge, Tennessee, 1965)

PARKER, H. M., AND NEWTON, C. E., JR. (1965). "The Hanford criticality accident: dosimetry techniques, interpretations and problems," *Personnel Dosimetry for Radiation Accidents*, p. 567 (Int. Atomic Energy Agency, Vienna, 1965)

PEABODY, C. O. (1963). *The response of some health physics instruments to sodium-24 and chlorine-38 activities in polythene manphantoms and the human body*, AEEW-M-335 (U.K. Atomic Energy Authority. Research Group. Atomic Energy Establishment, Winfrith, Dorset, England, 1963)

PETERSEN, D. F. (1963). "Neutron dose estimates in the SL-1 accident," Health Physics **9**, 231 (1963)

PETERSEN, D. F. (1965). "Rapid estimation of fast neutron doses following radiation exposure in criticality accidents: the ^{32}S (n,p)^{32}P reaction in body hair," *Personnel Dosimetry for Radiation Accidents,* p. 217 (Int. Atomic Energy Agency, Vienna, 1965)

PETERSEN, D. F., MITCHELL, V. E. AND LANGHAM, W. H. (1961). "Estimation of fast neutron doses in man by ^{32}S (n,p)^{32}P reaction in body hair," Health Physics **6,** 1 (1961)

PORAT, D. I. AND RAMAVATARAN, K. (1961). "Rate of energy loss and ranges of carbon and oxygen ions in solids," Proc. Phys. Soc. (London) **77,** 97 (1961)

PRICE, B. T., HORTON, C. C. AND SPINNEY, K. T. (1957). *Radiation shielding.* (Pergamon Press, New York, 1957)

PRICE, W. J. (1958). *Nuclear Radiation Detection.* (McGraw-Hill Book Company, New York, 1958)

REINIG, W. C., WRIGHT, C. N. AND HOY, J. E. (1965). "A pocket dosimeter for criticality accidents," *Personnel Dosimetry for Radiation Accidents,* p. 307 (Int. Atomic Energy Agency, Vienna, 1965)

RITCHIE, R. H. AND ELDRIDGE, H. B. (1961). "Calculation of the radiation yield from fission assemblies and comparison with experiments," *Selected Topics in Radiation Dosimetry,* p. 657 (Int. Atomic Energy Agency, Vienna, 1961)

ROCKWELL, T. (1956). *Reactor Shielding Design Manual.* (Office of Technical Services, Dept. of Commerce, Washington, 1956)

ROSSI, H. H., ROSENZWEIG, B., BIAVATI, M. H., GOODMAN, L. AND PHILLIPS, L. (1962). "Radiation protection surveys at heavy-particle accelerators operating at energies beyond several hundred million electron-volts," Health Physics **8,** 331 (1962)

SANDERS, F. W. AND AUXIER, J. A. (1962). "Neutron activation of sodium in anthropomorphous phantoms," Health Physics **8,** 371 (1962)

SANNA, R. AND O'BRIEN, K. (1963). "The use of nuclear emulsions to estimate stray neutron spectra near particle accelerators," Health Physics **9,** 25 (1963)

SAYRES, A. AND COPPOLA, M. (1964). "^{3}He neutron spectrometer using pulse risetime discrimination," Rev. Sci. Inst. **35,** 431 (1964)

SCHMIDT, F. A. R. (1968). *Revised Neutron Kerma Values at the boundary of a semi-infinite medium,* Report ORNL-TM-2284, (Oak Ridge National Laboratory, Oak Ridge, Tennessee, 1968)

SMITH, J. W. (1962). "Sodium activation by fast neutrons in man phantoms," Phys. in Med. and Biol. **7,** 341 (1962)

SMITH, J. W., BOOT, S. J., AND DENNIS, J. A. (1965). "A criticality dose assessment system," *Personnel Dosimetry for Radiation Accidents,* p. 369 (Int. Atomic Energy Agency, Vienna, 1965)

SNYDER, W. S. (1964). "The LET distribution of dose in some tissue cylinders," *Biological Effects of Neutron and Proton Irradiations,* Vol. I, p. 3 (Int. Atomic Energy Agency, Vienna, 1964)

THURSTON, M. O., SWARTZ, J. M., SPEERS, R. R. AND CLOSSER, W. H. (1967).

"A silicon fast-neutron dosimeter with a wide sensitivity range," *Neutron Monitoring*, p. 245 (Int. Atomic Energy Agency, Vienna, 1967)

TIPTON, I. H., JOHNS, J. C. AND BOYD, M. (1969). *Variation with age of elemental concentrations in human tissue.* ORNL-4446, (Oak Ridge National Laboratory, Tennessee, 1969). Also in *Radiation Protection*, Part I, p. 759, Snyder, W. S., Abee, H. H., Burton, L. K., Maushart, R., Benco, A., Duhamel, F. and Wheatley, B. M., eds. (Pergamon Press, Inc., New York, 1968)

TRUBEY, D. K. AND EMMETT, M. B. (1965). *Some calculations of the fast neutron distribution in ordinary concrete from point and plane isotropic fission sources*, ORNL-RSIC-4 (Oak Ridge National Laboratory, Oak Ridge, Tennessee, 1965)

UCNC (1958). *Accidental radiation excursion at the Y-12 plant, June 16, 1958.* Y-1234 (Union Carbide Nuclear Company, Oak Ridge, Tennessee, 1958)

UNRUH, C. M., BAUMGARTNER, W. V., KOCHER, L. F., BRACKENBUSH, L. W. AND ENDRES, G. W. R. (1967). "Personnel neutron dosimeter developments," *Neutron Monitoring*, p. 433 (Int. Atomic Energy Agency, Vienna, 1967)

USAEC (1966). Report CONF-651109, *Proceedings of the USAEC first symposium on accelerator radiation dosimetry and experience*, Brookhaven National Laboratory, Upton, New York, Nov. 3–5, 1965 (U.S. Atomic Energy Commission, Washington, 1966)

WACHTER, J. W. AND EMERSON, L. C. (1956). *Neutron dose calibration of indium personnel dosimeters for prompt-critical metal bursts*, Y-1092 (Carbide Nuclear Company, Oak Ridge, Tennessee, 1956)

WAGNER, E. G. AND HURST, G. S. (1961). "A Geiger-Mueller γ-ray dosimeter with low neutron sensitivity," Health Physics **5,** 20 (1961)

WILSON, R. H. (1962). *A method of immediate detection of high-level neutron exposure by measurement of sodium-24 in humans*, HW-73891 Rev. (General Electric Co., Hanford Atomic Products Operation, Richland, Washington, 1962)

ZERBEY, C. D. AND KINNEY, W. E. (1965). "Calculated tissue current-to-dose conversion factors for nucleons below 400 MeV," Nucl. Inst. Methods **36,** 125 (1965), also as ORNL-TM-1038 (Oak Ridge National Laboratory, Oak Ridge, Tennessee, 1965)

General References

AALTO, E., "Comparison of measured and calculated neutron fluxes in laminated iron and heavy water," Nucl. Sci. Eng. **22,** 94 (1965)

ARONSON, R., CERTAINE, J. AND GOLDSTEIN, H., *Penetration of neutrons from a point isotopic fission source in water*, AEC Report NYO 6267 (New York Operations Office, U.S. Atomic Energy Commission, New York, 1954)

ARONSON, R., CERTAINE, J. AND GOLDSTEIN, H., *Penetration of neutrons from point isotropic monoenergetic sources in water*, AEC Report NYO 6269

(New York Operations Office, U.S. Atomic Energy Commission, New York, 1954)

BARSUKOV, O. A. AND AVZYANOV, V. S., "Metal-water shielding from point neutron sources," J. Nucl. Energy, Pts. A/B. **19,** 114 (1965)

BEISSNER, R. E., *An analysis of fast neutron energy angle distributions,* FZK-9-186 (General Dynamics, Ft. Worth, Texas, 1963)

BRODER, D. L., ZAITSEV, L. N., SYCHEV, B. S. AND TUGOLOV, A. M., "Effect of water content in concrete on thickness and cost of reactor shielding," Soviet J. At. Energy (English Translation) **16,** 26 (1964)

CASPER, A. W., *Comparison of bulk shielding reactor centerline measurements in water with predictions,* Apex-504 (General Electric Aircraft Nuclear Propulsions, Cincinnati, Ohio, 1958)

ERMAKOV, S. M., ZOLOPTUKHIN, C. G. AND PETROV, E. E., "Calculation of the penetration of neutrons through a plane layer of polyethylene," ORNL-tr-(1964), Soviet J. At. Energy (English Translation), **15,** 960 (1963)

GOLDBERG, M. D., MAY, V. M. AND STEHN, J. R., *Angular distributions in neutron-induced reactions,* Vols. I and II, BNL-400 (Brookhaven National Laboratory, Upton, Long Island, New York, 1962)

HENRY, R. L., MOONEY, L. G. AND PREVOST, R. J., *Study of radiation penetration and reflection from shield materials,* FZK-183 (General Dynamics, Ft. Worth, Texas, 1964)

HJARNE, L. AND LEIMDORFER, M., "A method for predicting the penetration and slowing down of neutrons in reactor shields," Nucl. Sci. Eng. **24,** 165 (1966)

JAEGER, R. G., editor, *Engineering compendium on radiation shielding* (Springer-Verlag, New York, 1968)

KRUMBEIN, A. D., *Summary of NDA unclassified results of moments calculations for the penetration of neutrons through various materials,* NDA 92-2 (Nuclear Development Corp. of America, White Plains, New York, 1958)

LEIMDORFER, M., "On the use of the Monte Carlo method for calculating the deep penetration of neutrons in shields," Chalmers Tekn. Hogsk. Handl. NR 287 (Gothenberg, Sweden, 1964)

SHURE, K., *Experimental verification of neutron attenuation kernels,* WAPD-BT-17 (Bettis Laboratory, Westinghouse Corp., Pittsburgh, Pa., 1960)

SHURE, K., "P-3 multigroup calculations of neutron attenuation," Nucl. Sci. Eng., **19,** 310 (1964)

SPIEGELMANN, W. H., *Radioisotope Shielding Design Manual,* NYO-10721 (New York Operations Office, U.S. Atomic Energy Commission, New York, 1963)

The NCRP

The National Council on Radiation Protection and Measurements is a nonprofit corporation chartered by Congress in 1964 to:

1. Collect, analyze, develop, and disseminate in the public interest information and recommendations about (a) protection against radiation and (b) radiation measurements, quantities, and units, particularly those concerned with radiation protection;
2. Provide a means by which organizations concerned with the scientific and related aspects of radiation protection and of radiation quantities, units, and measurements may cooperate for effective utilization of their combined resources, and to stimulate the work of such organizations;
3. Develop basic concepts about radiation quantities, units, and measurements, about the application of these concepts, and about radiation protection.
4. Cooperate with the International Commission on Radiological Protection, the Federal Radiation Council, the International Commission on Radiation Units and Measurements, and other national and international organizations, governmental and private, concerned with radiation quantities, units, and measurements and with radiation protection.

The Council is the successor to the unincorporated association of scientists known as the National Committee on Radiation Protection and Measurements and was formed to carry on the work begun by the Committee.

The Council is made up of the members and the participants who serve on the thirty-four Scientific Committees of the Council. The Scientific Committees, composed of experts having detailed knowledge and competence in the particular area of the Committee's interest, draft proposed recommendations. These are then submitted to the full membership of the Council for careful review and approval before being published.

The following comprise the current officers and membership of the Council:

Officers

President	LAURISTON S. TAYLOR
Vice President	RUSSELL H. MORGAN
Secretary and Treasurer	W. ROGER NEY
Assistant Secretary	JOHN A. LODGE
Assistant Treasurer	HAROLD O. WYCKOFF

Members

S. ABRAHAMSON
E. A. ALPEN
E. C. BARNES
M. A. BENDER
V. P. BOND
F. J. BONTE
C. B. BRAESTRUP
J. T. BRENNAN
R. F. BROWN
A. M. BRUES
W. W. BURR, JR.
L. K. BUSTAD
G. W. CASARETT
R. S. CASWELL
R. H. CHAMBERLAIN
C. L. COMAR
F. P. COWAN
C. L. DUNHAM
M. EISENBUD
T. S. ELY
R. D. EVANS
T. C. EVANS
D. C. FLECKENSTEIN
R. C. FOSTER
R. O. GORSON
W. T. HAM, JR.
J. W. HEALY
J. M. HESLEP
J. S. LAUGHLIN
G. V. LEROY
C. W. MAYS
D. W. MOELLER
K. Z. MORGAN
R. H. MORGAN
R. D. MOSELEY, JR.
R. J. NELSEN
E. F. OAKBERG
H. M. PARKER
E. H. QUIMBY
A. RAVENTOS
W. C. ROESCH
H. H. ROSSI
W. L. RUSSELL
J. H. RUST
E. L. SAENGER
H. F. SCHULTE
F. J. SHORE
W. K. SINCLAIR
J. N. STANNARD
J. R. STORER
L. S. TAYLOR
J. G. TERRILL, JR.
J. H. TOLAN
E. D. TROUT
B. F. TRUM
A. C. UPTON
N. WALD
S. WARREN
E. W. WEBSTER
F. WESTERN
C. M. WILLIAMS
E. G. WILLIAMS
M. D. WILLIAMS
H. O. WYCKOFF

Honorary Member

P. C. HODGES

Currently, the following Scientific Committees are actively engaged in formulating recommendations:

SC-1: Basic Radiation Protection Criteria
SC-7: Monitoring Methods and Instruments
SC-11: Incineration of Radioactive Waste

SC-14: Radiological Factors Affecting Decision Making in a Nuclear Attack
SC-22: Radiation Shielding for Particle Accelerators
SC-23: Radiation Hazards Resulting from the Release of Radionuclides into the Environment
SC-24: Radionuclides and Labeled Organic Compounds Incorporated in Genetic Material
SC-25: Radiation Protection in the Use of Small Neutron Generators
SC-26: High Energy X-Ray Dosimetry
SC-27: Specification of Radium Substitutes
SC-28: Radiation Exposure from Consumer Products
SC-29: Brachytherapy
SC-30: Physical and Biological Properties of Radionuclides
SC-31: Selected Occupational Exposure Problems Arising from Internal Emitters
SC-32: Administered Radioactivity
SC-33: Dose Calculations
SC-34: Maximum Permissible Concentrations for Occupational and Non-Occupational Exposures

In recognition of its responsibility to facilitate and stimulate cooperation among organizations concerned with the scientific and related aspects of radiation protection and measurement, the Council has created a category of NCRP Collaborating Organizations. Organizations or groups of organizations which are national or international in scope and are concerned with scientific problems involving radiation quantities, units, measurements and effects, or radiation protection may be admitted to collaborating status by the Council. The present Collaborating Organizations with which the NCRP maintains liaison are as follows:

American Academy of Dermatology
American Association of Physicists in Medicine
American College of Radiology
American Dental Association
American Industrial Hygiene Association
American Insurance Association
American Medical Association
American Nuclear Society
American Public Health Association
American Radium Society
American Roentgen Ray Society
American Society of Radiologic Technologists
American Veterinary Medical Association
Association of University Radiologists
Atomic Industrial Forum
Genetics Society of America
Health Physics Society
Industrial Medical Association
National Bureau of Standards
National Electrical Manufacturers Association

Radiation Research Society
Radiological Society of North America
Society of Nuclear Medicine
United States Air Force
United States Army
United States Atomic Energy Commission
United States Navy
United States Public Health Service

The NCRP has found its relationships with these organizations to be extremely valuable to continued progress in its program.

The Council's activities are made possible by the voluntary contribution of the time and effort of its members and participants and the generous support of the following organizations:

Alfred P. Sloan Foundation
American Academy of Dental Radiology
American Academy of Dermatology
American Association of Physicists in Medicine
American College of Radiology
American College of Radiology Foundation
American Dental Association
American Industrial Hygiene Association
American Medical Association
American Mutual Insurance Alliance
American Nuclear Society
American Public Health Association
American Radium Society
American Roentgen Ray Society
American Society of Radiologic Technologists
American Veterinary Medical Association
American Veterinary Radiology Society
Association of University Radiologists
Atomic Industrial Forum
College of American Pathologists
Edward Mallinckrodt, Jr. Foundation
Federal Radiation Council
Genetics Society of America
Health Physics Society
Industrial Medical Association
National Bureau of Standards
National Electrical Manufacturers Association
Office of Civil Defense
Radiation Research Society
Radiological Society of North America
Society of Nuclear Medicine
United States Atomic Energy Commission
United States Public Health Service

To all of these organizations the Council expresses its profound appreciation for their support.

Initial funds for publication of NCRP reports are being provided by a grant from the James Picker Foundation and for this the Council wishes to express its deep appreciation.

The NCRP seeks to promulgate information and recommendations based on leading scientific judgment on matters of radiation protection and measurement and to foster cooperation among organizations concerned with these matters. These efforts are intended to serve the public interest and the Council welcomes comments and suggestions on its reports or activities from those interested in its work.

NCRP Reports

The NCRP Reports listed below are available from:
NCRP Publications
P. O. Box 4867
Washington, D. C. 20008

NCRP Report No.	Title	Price*
8	*Control and Removal of Radioactive Contamination in Laboratories* (1951)	$1.00
9	*Recommendations for Waste Disposal of Phosphorus-32 and Iodine-131 for Medical Users* (1951)	$1.00
10	*Radiological Monitoring Methods and Instruments* (1952)	$1.00
12	*Recommendations for the Disposal of Carbon-14 Wastes* (1953)	$1.00
14	*Protection Against Betatron-Synchroton Radiations Up To 100 Million Electron Volts* (1954).	$1.00
16	*Radioactive Waste Disposal in the Ocean* (1954).	$1.00
17	*Permissible Dose from External Sources of Ionizing Radiation* (1954) [Includes Addendum to NBS Handbook 59 issued on April 15, 1958]	$1.00
22	*Maximum Permissible Body Burdens and Maximum Permissible Concentrations of Radionuclides in Air and in Water for Occupational Exposure* (1959) [Includes Addendum 1 to NBS Handbook 69 issued in August 1963].	$1.00
23	*Measurement of Neutron Flux and Spectra for Physical and Biological Applications* (1960).	$1.00
24	*Protection Against Radiations from Sealed Gamma Sources* (1960)	$1.00
25	*Measurement of Absorbed Dose of Neutrons and of Mixtures of Neutrons and Gamma Rays* (1961)	$1.00

* The prices listed include postage. Orders should be accompanied by a check or money order, but NCRP Publications will invoice purchases if necessary. Prices subject to change without notice.

NCRP Report No.	Title	Price*
27	*Stopping Powers for Use with Cavity Chambers* (1961)	$1.00
28	*A Manual of Radioactivity Procedures* (1961)	$1.00
29	*Exposure to Radiation in an Emergency* (1962)	
	1 through 99 copies, each copy	$1.00
	100 through 999 copies, each copy	$0.85
	1,000 or more copies, each copy	$0.75
30	*Safe Handling of Radioactive Materials* (1964)	$1.00
31	*Shielding for High-Energy Electron Accelerator Installations* (1964)	$1.00
32	*Radiation Protection in Educational Institutions* (1966)	
	1 through 99 copies, each copy	$1.00
	100 or more copies, each copy	$0.85
33	*Medical X-Ray and Gamma-Ray Protection for Energies up to 10 MeV—Equipment Design and Use* (1968)	
	1 through 99 copies, each copy	$1.00
	100 or more copies, each copy	$0.85
34	*Medical X-Ray and Gamma-Ray Protection for Energies up to 10 MeV—Structural Shielding Design and Evaluation* (1970)	
	1 through 99 copies, each copy	$1.50
	100 or more copies, each copy	$1.25
35	*Dental X-Ray Protection* (1970)	
	1 through 99 copies, each copy	$1.50
	100 or more copies, each copy	$1.25
36	*Radiation Protection in Veterinary Medicine* (1970)	
	1 through 99 copies, each copy	$1.50
	100 or more copies, each copy	$1.25
37	*Precautions in the Management of Patients Who Have Received Therapeutic Amounts of Radionuclides* (1970)	
	1 through 99 copies, each copy	$1.50
	100 or more copies, each copy	$1.25
38	*Protection Against Neutron Radiation* (1971)	
	1 through 99 copies, each copy	$3.00
	100 or more copies, each copy	$2.75

* The prices listed include postage. Orders should be accompanied by a check or money order, but NCRP Publications will invoice purchases if necessary. Prices subject to change without notice.

The following NCRP reports are now superseded and/or out of print:

NCRP Report No.	Title
1	*X-Ray Protection* (1931). [Superseded by NCRP Report No. 3]
2	*Radium Protection* (1934). [Superseded by NCRP Report No. 4]
3	*X-Ray Protection* (1936). [Superseded by NCRP Report No. 6]
4	*Radium Protection* (1938). [Superseded by NCRP Report No. 13]
5	*Safe Handling of Radioactive Luminous Compounds* (1941). [Out of print]
6	*Medical X-Ray Protection up to Two Million Volts* (1949). [Superseded by NCRP Report No. 18]
7	*Safe Handling of Radioactive Isotopes* (1949). [Superseded by NCRP Report No. 30]
11	*Maximum Permissible Amounts of Radioisotopes in the Human Body and Maximum Permissible Concentrations in Air and Water* (1953). [Superseded by NCRP Report No. 22]
13	*Protection Against Radiations from Radium, Cobalt-60 and Cesium-137* (1954). [Superseded by NCRP Report No. 24]
15	*Safe Handling of Cadavers Containing Radioactive Isotopes* (1953). [Superseded by NCRP Report No. 21]
18	*X-Ray Protection* (1955). [Superseded by NCRP Report No. 26]
19	*Regulation of Radiation Exposure by Legislative Means* (1955). [Out of print]
20	*Protection Against Neutron Radiation Up to 30 Million Electron Volts* (1957). [Superseded by NCRP Report No. 38]
21	*Safe Handling of Bodies Containing Radioactive Isotopes* (1958). [Superseded by NCRP Report No. 37]
26	*Medical X-Ray Protection Up to Three Million Volts* (1961). [Superseded by NCRP Reports Nos. 33, 34, 35 and 36]

The following statements of the NCRP were published outside of the NCRP Reports series:

"Blood Counts, Statement of the National Committee on Radiation Protection," Radiology **63,** 428 (1954)

"Statements on Maximum Permissible Dose from Television Receivers and Maximum Permissible Dose to the Skin of the Whole Body," Am. Jr. of Roentgenol., Radium Therapy and Nucl. Med. **84,** 152 (1960) and Radiology **75,** 122 (1960)

X-Ray Protection Standards for Home Television Receivers, Interim Statement of the National Council on Radiation Protection and Measurements (National Council on Radiation Protection and Measurements, Washington, 1968)

Copies of the statements published in journals may be consulted in libraries. A limited number of copies of the last statement listed above are available for distribution by NCRP Publications.

Index*

* The NCRP is greatly indebted to W. S. Snyder, D. M. Davis and T. J. Burnett for their assistance in connection with the preparation of this index.